FORM-14159 R001*

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Bruce Power

Bruce Power Document #:	Revision:	Information Classification	Usage Classification
B-GAR-09701-00001	R002	Internal Use Only	Information

Bruce Power Document Title:
Bruce A and B Global Assessment Report and Integrated Implementation Plan

Bruce Power Contract/Purchase Order:	Bruce Power Project #:
00193829/00219454	38180/39075

Supplier's Name:	Supplier Document #:	Revision:
CANDESCO	K-421231-00217	R02

Supplier Document Title:
Bruce A and B Global Assessment Report and Integrated Implementation Plan

Accepted for use at Bruce Power by:	Signature:	Date
Name: Mike Rencheck Title: President and CEO	Michael W. Rencheck	July 19, 2017

Acceptance of this document does not relieve the Supplier of responsibility for any errors or omissions

*Associated with BP-PROC-01024, Periodic Safety Reviews

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Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

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Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

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- **P. BOUCHER**
- **G. NEWMAN**
- **P. WILSON**
- **C. HORTON**
- **D. ANDREWS**
- **P. CLARK**
- **P. PILON**
- **J. SCONGACK**
- **J. PHELPS**
- **C. TRAHAN**

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Bruce Power Contract/ Purchase Order:	00193829/00219454	Supplier Document Title:	Bruce A and B Global Assessment Reort and Integrated Implementation Plan	
Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

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Bruce Power Document Title:	Bruce A and B Global Assessment Reort and Integrated Implementation Plan	Suppliers Name:	CANDESCO	
Bruce Power Contract/ Purchase Order:	00193829/00219454	Supplier Document Title:	Bruce A and B Global Assessment Reort and Integrated Implementation Plan	
Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

Reviewed By:

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Bruce Power Project #:	38180/39075	**Supplier Document:**	K-421231-00217	Rev #: R02

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Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

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Bruce Power Contract/ Purchase Order:	00193829/00219454	**Supplier Document Title:**	Bruce A and B Global Assessment Reort and Integrated Implementation Plan	
Bruce Power Project #:	38180/39075	**Supplier Document:**	K-421231-00217	**Rev #: R02**

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Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02	

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Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

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Bruce Power Contract/ Purchase Order:	00193829/00219454	Supplier Document Title:	Bruce A and B Global Assessment Reort and Integrated Implementation Plan	
Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02

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Bruce Power Contract/ Purchase Order:	00193829/00219454	Supplier Document Title:	Bruce A and B Global Assessment Reort and Integrated Implementation Plan		
Bruce Power Project #:	38180/39075	Supplier Document:	K-421231-00217	Rev #: R02	

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Chris Trahan	VP	PMC		17 July 2017

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Division of Kinectrics Inc.

Title: Bruce A and B Global Assessment Report and Integrated Implementation Plan

File: K-421231-00217-R02

B-GAR-09701-00001

R002

A Report Submitted to Bruce Power

July 7, 2017

<table>
<tr><td rowspan="2">Issue
-00026-R00D1</td><td colspan="5">Reason for Issue:
Bruce A stand-alone GAR/IIP (K-421231-00026-R00D1): Issued for internal review</td></tr>
<tr><td>Author:
T. Kapaklili</td><td>Verifier:</td><td>Reviewer:
G. Archinoff
L. Watt</td><td>Approver:</td><td>Date:
Sep 14, 2015</td></tr>
<tr><td rowspan="2">Issue
-00026-R00D2</td><td colspan="5">Reason for Issue:
Bruce A stand-alone GAR/IIP (K-421231-00026-R00D1): Comments from review of K-421231-00026-R00D1 addressed; for further internal review.</td></tr>
<tr><td>Author:
T. Kapaklili</td><td>Verifier:</td><td>Reviewer:
M. Trandafirescu</td><td>Approver:</td><td>Date:
Oct 6, 2015</td></tr>
<tr><td rowspan="2">Issue
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Bruce A stand-alone GAR/IIP (K-421231-00026-R00D1): For Bruce Power review</td></tr>
<tr><td>Author:
T. Kapaklili</td><td>Verifier:</td><td>Reviewer:
G. Archinoff
L. Watt
M. Trandafirescu</td><td>Approver:</td><td>Date:
Oct 19, 2015</td></tr>
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Bruce A and B GAR/IIP: For initial acceptance of tracked changes in combining Bruce A and B GAR/IIP</td></tr>
<tr><td>Author:
T. Kapaklili</td><td>Verifier:</td><td>Reviewer:
L. Watt</td><td>Approver:</td><td>Date:
July 18, 2016</td></tr>
</table>

Issue R00D1	**Reason for Issue:** Bruce A and B GAR/IIP: Reformatted version				
	Author: T. Kapaklili	Verifier:	Reviewer: L. Watt	Approver:	Date: July 21, 2016
Issue R00D2	**Reason for Issue:** For internal review				
	Author: T. Kapaklili	Verifier:	Reviewer: G. Archinoff L. Watt	Approver:	Date: Sept. 2, 2016
Issue R00D3	**Reason for Issue:** For Bruce Power review				
	Author: T. Kapaklili	Verifier:	Reviewer: G. Archinoff L. Watt	Approver:	Date: Sept. 26, 2016
Issue R00D4	**Reason for Issue:** Incorporates dispositions to Bruce Power comments				
	Author: T. Kapaklili	Verifier:	Reviewer: G. Archinoff L. Watt	Approver:	Date: Nov. 17, 2016

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For use</td></tr>
<tr><td>Author:
T. Kapaklili
H. Semeralul
C. Gillis</td><td>Verifier:
G. Buckley</td><td>Reviewer:
G. Archinoff
L. Watt</td><td>Approver:
L. Watt</td><td>Date:
Nov. 22, 2016</td></tr>
<tr><td rowspan="2">Issue
R01D1</td><td colspan="5">Reason for Issue:
Alert Groups augmented and commitment dates refined, plus minor corrections made to document</td></tr>
<tr><td>Author:
T. Kapaklili
H. Semeralul
C. Gillis</td><td>Verifier:</td><td>Reviewer:
G. Archinoff
L. Watt</td><td>Approver:</td><td>Date:
Dec. 2, 2016</td></tr>
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For use</td></tr>
<tr><td>Author:
T. Kapaklili
H. Semeralul
C. Gillis</td><td>Verifier:
R. Gold</td><td>Reviewer:
G. Archinoff
L. Watt</td><td>Approver:
L. Watt</td><td>Date:
Dec. 9, 2016</td></tr>
<tr><td rowspan="2">Issue
R02D0</td><td colspan="5">Reason for Issue:
For Bruce Power review. The most significant modifications are: IIP initiatives augmented; Alert Group, Functional Area and Target Commitment Dates refined; and CNSC feedback on Revision R01 incorporated.</td></tr>
<tr><td>Author:
T. Kapaklili
C. Gillis</td><td>Verifier:</td><td>Reviewer:
L. Watt</td><td>Approver:</td><td>Date:
May 23, 2017</td></tr>
</table>

candesco Division of Kinectrics Inc.	Rev Date: July 7, 2017	Status: Issued
	Subject: Bruce A and B Global Assessment Report and Integrated Implementation Plan	File: K-421231-00217-R02

Issue	Reason for Issue:				
R02D1	For internal review. Further CNSC feedback on defence-in-depth incorporated.				
	Author: T. Kapaklili C. Gillis P. Ardenska D. Duncan	Verifier:	Reviewer: G. Archinoff L. Watt	Approver:	Date: June 22, 2017

Issue	Reason for Issue:				
R02D2	For Bruce Power review				
	Author: T. Kapaklili C. Gillis P. Ardenska D. Duncan	Verifier: J. Dinner	Reviewer: G. Archinoff L. Watt	Approver:	Date: June 27, 2017

Issue	Reason for Issue:				
R02	For use				
	Author: T. Kapaklili C. Gillis P. Ardenska D. Duncan for D.D.	Verifier: J. Dinner	Reviewer: G. Archinoff L. Watt	Approver: L. Watt	Date: July 7, 2017

Document Classification:	Security Classification:
Report	Client Proprietary

Executive Summary

This Global Assessment Report (GAR) / Integrated Implementation Plan (IIP) represents the results of a Periodic Safety Review (PSR) of Bruce A and Bruce B. The PSR has been carried out in accordance with the Bruce A ISR (Integrated Safety Review) Basis Document and Bruce B PSR Basis Document as described in [1] and [2], which were accepted by the CNSC for implementation in [3] and [4].

Bruce Power has developed an asset management strategy and plan in support of life extension of Units 1 to 8 as an essential part of their continued safe and reliable operation. This asset management strategy is being integrated with a PSR process as identified in Section 15.2 of the Power Reactor Operating Licence (PROL) [5][1] to ensure compliance with the current licensing basis and to assess the plant against modern codes and standards and to implement practicable improvements to enhance plant safety. Figure 1 illustrates the overall framework which has been put in place. The PSR will be updated nominally every 10 years. This framework allows for updates of the IIP periodically and its rebaselining approximately every 10 years through the PSR process.

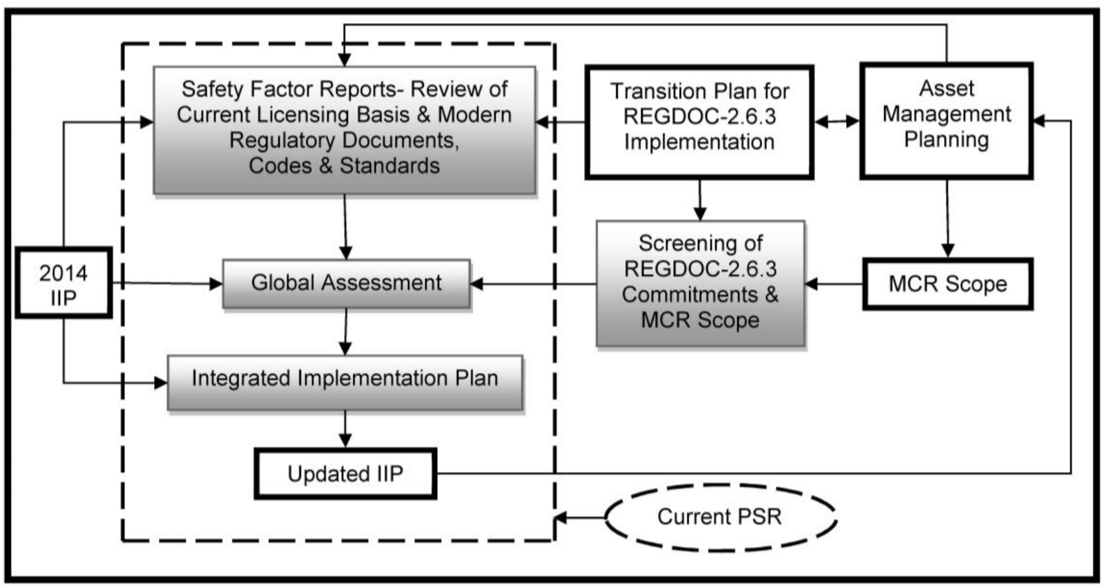

Figure 1: Bruce Power's Framework for Safe and Reliable Operation

[1] PROL 18.00/2020 [5] and LCH-BNGS-R000 [6] came into effect on June 1, 2015, and were used for the Bruce B PSR. However, PROL 15.00/2015 [7] and LCH-BNGSA-R8 [8] are the versions referred to in the Bruce A ISR, as these were in force when the assessments in the Bruce A Safety Factor Reports (SFRs) were prepared.

As part of the asset management strategy and the Government of Ontario's Long Term Energy Plan (LTEP), Bruce Power is planning to continue operation of Bruce A and Bruce B well into the future. For Units 3 to 8, this will require replacing major components such as pressure tubes, feeders and steam generators as part of Bruce Power's Asset Management Plan (AMP). The AMP activities will be executed before, during and after the Major Component Replacement (MCR). The alignment of the MCR and asset management plans will support Bruce Power's goal for safe long term operation. The MCR approach has been put in place and communicated to the CNSC [9] [10] to replace major components over the life of the units.

Bruce Power intends to submit a PSR based licence application in Q3 2017, to include the IIP into the licensing basis consistent with Licence Conditions Handbook Section 15.2. The licence application will include a formal notice of intent for the first Bruce B unit (Unit 6) MCR outage program, a project execution plan and a return to service plan, in accordance with PROL 18.00/2020 Licence Condition 15.2 and Licence Conditions Handbook Section 15.2. The conduct of this PSR supports the definition and timing of practicable opportunities for enhancing the safety of Units 3 to 8 and allows for integration of MCR scope into the PSR process and the IIP. This PSR is also applicable to the ongoing operation of Units 1 and 2, which have already been refurbished.

This PSR builds upon the previous ISRs conducted. In general, this PSR and subsequent PSRs will focus on changes in requirements, facility conditions, operating experience and new information, rather than repeating the activities of previous reviews.

The PSR has been conducted in a systematic and comprehensive manner across 15 Safety Factors in accordance with PSR basis documents accepted by the CNSC in [3] and [4]. The objectives of PSR are to:

- Determine the extent to which the plant meets modern codes and standards and industry best practices;
- Determine the extent to which the licensing basis will remain valid over the operating life of Bruce A and Bruce B;
- Determine the adequacy of the Structures, Systems, and Components (SSCs) and programs that are in place to ensure plant safety for long-term operation; and
- Determine the practicable improvements to be implemented to resolve any findings identified in the review and timelines for their implementation

As described above, the PSR includes a review against modern codes and standards, but not a requirement to meet modern codes and standards. The review against modern codes and standards facilitates identification of opportunities for nuclear safety improvements that can be practicably implemented over the 10-year timeline of the PSR. This review also serves to demonstrate Bruce Power's commitment to continuously improve safe and reliable operation and maintain a strong nuclear safety culture.

A Global Assessment of the results of the PSR has been completed using a systematic methodology. The GA methodology covers the following elements:

- Development of a Global Assessment Framework (GAF) which is a common methodology and basis for systematically assessing the relative importance of diverse PSR findings in terms of their safety significance and impact of their resolution. The same framework is used to assess the efficacy of practicable improvements and associated corrective actions for the resolution of findings;

- Integration of the results of the Safety Factor Reports (SFRs), in particular, the findings (both micro-gaps and strengths) in terms of overlaps, omissions, and interface issues;
- Assessment of interdependencies amongst integrated SFR micro-gaps, their classification in terms of practicability, consolidation into global issues and their safety significance;
- Definition of potential improvement opportunities and identification of improvement actions to achieve these safety improvements;
- Integration of improvement opportunities resulting from the GA and on-going improvement initiatives, such as those associated with the MCR outage(s) planned within this PSR period, into an IIP;
- Assessment of the extent to which the safety requirements of defence-in-depth are fulfilled;
- Estimation of global risk associated with facility operation with any unresolved gaps;
- A statement of the overall safety associated with facility operation over the applicable period of this PSR, taking into account the planned improvements as well as the impact of improvement opportunities that will not be implemented; and
- A final report (this report) summarizing the results of SFRs, and documenting the GA and the associated IIP.

All PSR findings have been consolidated, classified, the resulting potential improvements grouped as a set of Global Improvement Opportunities (GIOs) and ranked for inclusion in the IIP using the Global Assessment Framework. The IIP has been prepared and has been integrated with the previous IIP submitted to the CNSC [11] in support of the current operating licence. Each initiative has been ranked in terms of priority, with 1 being the highest priority. A high level list of the GIOs included in the IIP, which also includes GIOs from [11], planned MCR activities that ensure compliance with Licence Condition 15.2 Continued Operations of the PROL [5], as well as initiatives that support the licence application in Q3 2017 is tabulated below. The IIP is included in Appendix A and further details on implementation of associated Corrective Actions (CAs) for each GIO are described in Appendix H. In developing this list, the PSR has not eliminated any potential nuclear safety improvement on the basis of cost alone.

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce A & B	GIO-025	Fitness for service	Perform R&D in support of fuel channel life cycle management initiatives	1
Bruce A	GIO-028	Fitness for service	Upgrade Emergency and Standby Power Supplies	1
Bruce Units 3-8	GIO-056	Fitness for service	Fuel Channel Replacement	1
Bruce Units 3-8	GIO-057	Fitness for service	Steam Generator Replacement	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce Units 3-8	GIO-058	Fitness for service	Feeder Replacement	1
Bruce Units 3-8	GIO-062	Fitness for service	PHT Pump Seal Bellows Replacement	1
Bruce Units 3&4	GIO-064	Fitness for service	Control Distribution Frame (CDF) Terminal Replacement	1
Bruce Units 3-8	GIO-070	Fitness for service	Air Operated Valves-Replacement	1
Bruce Units 3-8	GIO-071	Fitness for service	Large Motors-Refurbishment/Replacement	1
Bruce Units 3-8	GIO-076	Fitness for service	DCC Cables and WIBAs –Replacement	1
Bruce Units 3-8	GIO-077	Fitness for service	Moderator Heat Exchangers- Replacement	1
Bruce Units 3-8	GIO-078	Fitness for service	Maintenance Cooling Heat Exchanger- Replacement	1
Bruce Units 3-8	GIO-086	Fitness for service	PHT Valves-Refurbishment of 33120-MV23	1
Bruce Units 3-8	GIO-095	Fitness for service	45VDC Power Supplies-Replacement	1
Bruce Units 3-8	GIO-100	Physical design	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications	1
Bruce A & B	GIO-101	Physical design	M/34720 Relief Valves For Overpressure Protection	1
Bruce A & B	GIO-102	Physical design	I/63472 Remote Relief Valve Position Indication	1
Bruce A & B	GIO-039	Fitness for service	Equipment Reliability and Maintenance	2
Bruce Units 3-8	GIO-059	Fitness for service	Calandria and Shield Tank Assembly Major Inspection	2
Bruce Units 3-8	GIO-060	Fitness for service	Preheater Inspections	2
Bruce Units 3-8	GIO-065	Fitness for service	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection	2
Bruce Units 3-8	GIO-066	Fitness for service	Pressurizer and Supports- Internal Inspection	2
Bruce A & B	GIO-104	Fitness for service	Ongoing Work on Bruce B Heat Transport Vibration Project	2
Bruce A	GIO-034	Fitness for service	Safety System Reliability	3
Bruce Units 1&2	GIO-019	Physical design	Assess and improve seismic qualification	4
Bruce A & B	GIO-082	Environmental protection	Performance testing of nuclear air-cleaning systems	5
Bruce A & B	GIO-089	Safety analysis	Whole-Site Probabilistic Risk Assessment	5

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce A & B	GIO-009	Safety analysis	Update safety analysis to align with REGDOC-2.4.1	6
Bruce A & B	GIO-103	Fitness for service	Implementation of Asset Management Activities	7
Bruce B	GIO-099	Physical design	Install Correctly Sized Maintenance Cooling Relief Valves	8
Bruce A & B	GIO-043	Human performance management	Validation of Human Credited Actions	9
Bruce A & B	GIO-093	Radiation protection	RP equipment and instrumentation maintenance and life cycle management	10
Bruce A & B	GIO-094	Radiation protection	Effective use of the action tracking system in Radiation Protection	11
Bruce A & B	GIO-011	Operating performance	Implement enhancements to SAMG	12
Bruce A & B	GIO-001	Physical design	Improve documented design basis	13
Bruce A & B	GIO-081	Physical design	Human Factors in Design of Nuclear Power Plants	13
Bruce A & B	GIO-083	Safety analysis	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2	13
Bruce A & B	GIO-044	Emergency management and fire protection	Emergency preparedness	14
Bruce A & B	GIO-088	Management system	Improve Licencing Processes	15
Bruce A & B	GIO-002	Physical design	Implement design changes to improve severe accident response	16
Bruce A & B	GIO-026	Physical design	BA & BB New Neutronic Trips	16
Bruce Units 3&4	GIO-090	Physical design	SDS2 Enhancements	16
Bruce A	GIO-091	Physical design	Bruce A Fire Protection Upgrades to Align with CSA-N293-07	16
Bruce B	GIO-092	Physical design	Bruce B Fire Protection Upgrades to Align with CSA-N293-07	16
Bruce A	GIO-097	Physical design	Bruce A Legacy Registration- Implementation Projects	16
Bruce B	GIO-098	Physical design	Bruce B Legacy Registration- Implementation Projects	16
Bruce A & B	GIO-024	Management system	Enhanced Periodic Safety Review to Support Asset Management	17
Bruce B	GIO-003	Physical design	Assess pipe whip and jet impingement	18
Bruce B	GIO-005	Physical design	Assess cyclic loads of pressure retaining components designed per ASME III or VIII	18

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce A & B	GIO-036	Physical design	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment	18
Bruce A & B	GIO-037	Physical design	Document design basis for zoning and shielding	18

As part of the Global Assessment, an evaluation of defence-in-depth (DID) and overall safety associated with plant design and operation has been conducted.

The overall conclusion of the Global Assessment is that continued operation of Bruce A and Bruce B over the designated PSR period is acceptable based on the results and conclusions of the GA and the commitment to implement the initiatives included in the IIP. This conclusion is based on the following:

- A comprehensive PSR of Bruce A and Bruce B has been completed. This review covered the current organization, governance and processes associated with all aspects of plant design, operation and condition of the physical plant against a set of review tasks identified in the PSR Basis Documents [1], [2], as well as modern codes and standards. No safety concerns requiring immediate action have been identified.
- The extent to which Bruce A and Bruce B currently meet new requirements that may become part of the licensing basis in the future has been assessed and practicable improvement opportunities have been included in the IIP. These improvement opportunities will further enhance safe and reliable operation and align Bruce A and Bruce B design and operation with modern regulatory documents, codes and standards applicable to new nuclear power plants (NPPs).
- Adequacy of the design and DID provisions in place has been demonstrated, based on review of the fundamental safety principles associated with all 5 levels of DID.
- Overall risk associated with operation of Bruce A and Bruce B over the designated PSR period is acceptably low. Design and operation of the plants meets the current deterministic safety analysis dose acceptance limits of PROL [5], as well as the probabilistic safety analysis safety goals with significant safety margins. The IIP implementation will further enhance the current safety basis of the plants to ensure that dose limits and risk goals are met over the PSR period and beyond.
- A framework as shown in Figure 1 is in place that integrates improvements planned or in-progress based on asset life management inputs and those proposed in the IIP to mitigate SSC aging for continued safe and reliable long-term operation.
- Bruce Power's current organizational structure and management system provides the requisite processes, tools, resources and oversight that will ensure effective execution of the IIP.
- Completion of the planned improvements in the IIP will enhance safe and reliable operation of Bruce A and Bruce B over the designated PSR period.

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Acronyms and Abbreviations

37M	Modified 37-element (Fuel Bundle)
AI	Action Item
AIM	Abnormal Incidents Manual
ALPO	Asset Life Projections and Options
AM	Asset Management
AMOT	Asset Management Options Template
AMP	Ageing Management Program
AOO	Anticipated Operational Occurrence
AOP	Ageing and Obsolescence Project
AOR	Analysis of Record
AR	Action Request
ARDM	Age Related Degradation Mechanism
ASDV	Atmospheric Steam Discharge Valve
ASME	American Society of Mechanical Engineers
B12PRA	Bruce 1 and 2 Probabilistic Risk Assessment
BA	Bruce A
BAPRA	Bruce A (3 and 4) Probabilistic Risk Assessment
BB	Bruce B
BBPRA	Bruce B Probabilistic Risk Assessment
BDBA	Beyond Design Basis Accident
BEAU	Best Estimate Analysis and Uncertainty
BERP	Bruce Emergency Response Projection
BP	Bruce Power
BPMS	Bruce Power Management System
BPVC	Boiler and Pressure Vessel Code
CA	Corrective Action
CALA	Canadian Association for Laboratory Accreditation
CANDU	CANada Deuterium Uranium
CARD	Corrective Action Requirements Definition

CARs	Condition Assessment Reports
CD	Cooldown
CFA	Central Fuelling Area
CFVS	Containment Filtered Venting System
CM	Configuration Management
CMF	Common Mode Failure
CMLF	Central Maintenance and Laundry Facility
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners' Group
COTS	Commercial Off The Shelf
Crit-Cat	Criticality Category
CSA	Canadian Standards Association
CSDV	Condenser Steam Discharge Valve
CSI	CANDU Safety Issue
CSP	Composite Safety Profile
CSTA	Calandria/Shield Tank Assembly
CT	Calandria Tube
DBA	Design Basis Accident
DBE	Design Basis Earthquake
DCC	Digital Control Computer
DCN	Design Change Notice
DCP	Design Change Package
DCR	Document Change Request
DEC	Design Extension Condition
DG	Design Guide
DID	Defence-in-Depth
DSA	Deterministic Safety Analysis
DSC	Dry Storage Container
EBC	Emergency Boiler Cooling
EC	Engineering Change
ECC	Engineering Change Control

ECI	Emergency Coolant Injection
EMC	Emergency Management Centre
EME	Emergency Mitigating Equipment
EMEG	Emergency Mitigating Equipment Guidance
EMO	Emergency Management Ontario
EOC	Emergency Operations Centre
EOL	End of Life
EOP	Emergency Operating Procedure
EPG	Emergency Power Generator
EPRI	Electric Power Research Institute
EPS	Emergency Protective Services
EQ	Environmental Qualification
ER	Equipment Reliability
ERO	Emergency Response Organization
EVP	Executive Vice President
FAC	Flow Accelerated Corrosion
FAI	Fukushima Action Item
FASA	Focus Area Self Assessment
FC	Fuel Channel
FFS	Fitness-for-Service
FSSA	Fire Safe Shutdown Assessment
GAF	Global Assessment Framework
GAI	General Action Item
GAR	Global Assessment Report
GD	Guidance Document
GI	Global Issue
GIO	Global Improvement Opportunity
HF	Human Factors
HMI	Human-Machine Interface
HTS	Heat Transport System
HX	Heat Exchanger (used to identify the Bruce Preheaters)

IAEA	International Atomic Energy Agency
IAMP	Integrated Accident Management Program
IFB	Irradiated Fuel Bay
IIP	Integrated Implementation Plan
INPO	Institute of Nuclear Power Operations
ISO	International Standards Organization
ISR	Integrated Safety Review
IUC	Instrument Uncertainty Calculation
JP	Joint Project
LBB	Leak-Before-Break
LCH	Licence Conditions Handbook
LCM	Life Cycle Management
LCMP	Life Cycle Management Plan
LISS	Liquid Injection Shutdown System
LLOCA	Large Loss of Coolant Accident
LOCA	Loss of Coolant Accident
LOE	Limit of Operating Envelope
LOF	Loss of Flow
LOR	Loss of Regulation
LRF	Large Release Frequency
LTEP	Long-Term Energy Plan
LWR	Light Water Reactor
MCG	Moderator Cover Gas
MCR	Major Component Replacement
MSM	Management System Manual
NBCC	National Building Code of Canada
NDE	Non-Destructive Examination
NERP	Nuclear Emergency Response Plan
NFCC	National Fire Code of Canada
NFPA	National Fire Protection Association
NOP	Neutron Overpower Protection

NORA	Nuclear Oversight and Regulatory Affairs
NPC	Negative Pressure Containment
NPP	Nuclear Power Plant
NSA	Nuclear Safety Assessment
NSAS	Nuclear Safety Analysis and Support
NSERC	Natural Sciences and Engineering Research Council of Canada
NuSCI	Nuclear Subject Classification Index
OHSA	Ontario Occupational Health and Safety Act
OLCs	Operating Limits and Conditions
OP&P	Operating Policies and Principles
OPEX	Operating Experience
OPG	Ontario Power Generation
OSR	Operational Safety Requirement
P&G	Principles & Guidelines
PAR	Passive Autocatalytic Recombiner
PARMS	Post-Accident Radiation Monitoring System
PassPort	Bruce Power’s data/information management system
PB	Pressure Boundary
PB QA	Pressure Boundary Quality Assurance
PDE	Plant Design Engineering
PEOC	Provincial Emergency Operations Centre
PH	Preheater
PHT	Primary Heat Transport
PHTS	Primary Heat Transport System
PI	Performance Indicator
PIE	Postulated Initiating Event
PIP	Periodic Inspection Plan (or Program)
PM	Predictive Maintenance
PM	Preventive Maintenance
PMOG	Preventive Maintenance Oversight Group
PRA	Probabilistic Risk Assessment (synonymous with PSA)

PROL	Power Reactor Operating Licence
PRV	Pressure Relief Valve
PSA	Probabilistic Safety Assessment (synonymous with PRA)
PSR	Periodic Safety Review
PT	Pressure Tube
PWR	Pressurized Water Reactor
QA	Quality Assurance
QPS	Qualified Power Supply
R&D	Research and Development
RCS	Regulations, Codes, and Standards
RD	Regulatory Document
RED	Radiation Emitting Devices
RIDM	Risk-Informed Decision Making
ROE	Realistic Operating Envelope
RS	Reactor Safety
RTS	Return to Service
SAI	Safety Analysis Improvement
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidance
SBO	Station Black Out
SBR	Safety Basis Report
SCA	Secondary Control Area
SCDF	Severe Core Damage Frequency
SCR	Station Condition Record
SDC	Shutdown Cooling
SDS	Shutdown System
SEA	Significant Environmental Aspect
SF	Safety Factor
SFR	Safety Factor Report
SG	Steam Generator
SHIP	System Health Improvement Plan

SHR	System Health Report
SIS	System Important to Safety
SLOCA	Small Loss of Coolant Accident
SMA	Seismic Margin Assessment
SMC	Site Management Centre
SME	Subject Matter Expert
SOE	Safe Operating Envelope
SOFA	State of the Functional Area
SOG	Standard Operating Guideline
SOW	Scope of Work
SP	Safety Principle
SPHC	Station Plant Health Committee
SPV	Single Point Vulnerability
SR	Safety Report
SRF	Small Release Frequency
SRI	Safety Report Improvement
SSCs	Structures, Systems, Components
SSCTs	Structures, Systems, Components and Significant Tools
SSMC	Safety System Monitoring Computer
SST	Safety System Test
TBA	Technical Basis Assessment
VP	Vice President
VSAT	Very Small Aperture Terminal
WANO	World Association of Nuclear Operators
WIBA	Weidmeuller Interface Board
WO	Work Order

Part I: Methodology

Section	Title
1	Purpose
2	Background
3	Overview of Global Assessment and IIP Process
4	Organization of the Report
5	Global Assessment Methodology

Appendix	Title
Appendix C	Global Assessment Framework

1. Purpose

This Global Assessment Report (GAR) / Integrated Implementation Plan (IIP) represents the results of a Periodic Safety Review (PSR) of Bruce A and Bruce B. The PSR includes a review against modern codes and standards, but not a requirement to meet modern codes and standards. The purpose of the review is to identify opportunities for nuclear safety improvements that can be practicably implemented over the ten year timeline of the PSR. This review also serves to demonstrate Bruce Power's commitment to continuously improve safe and reliable operation and maintain a strong nuclear safety culture.

The overall objective of the global assessment is to present an integrated evaluation of the facility safety and those in place for new nuclear power plants (NPPs) in modern codes and standards taking into account a balanced assessment of all findings identified in the Safety Factor Reports (SFRs). The global assessment takes into account all the strengths and micro-gaps from the SFRs, and the practicable corrective actions and/or safety improvements proposed in the IIP to provide an overall assessment of the safety of plant and acceptability of long-term operation.

The purpose of Part I of this GAR is to describe the methodology to implement the PSR process systematically as described in [1] and [2] that includes the following:

- Description of the process and framework used for Global Assessment (GA) and IIP development;
- A summary of the outcomes from the SFRs, including a list of findings indicating areas where the standards and practices considered in the PSR are not achieved, and a list of areas where they are exceeded (that is, plant strengths);
- A summary of the outcomes from the global assessment; and
- An IIP of practicable safety improvements, including their safety significance and prioritization.

The objective of the IIP is to provide the roadmap for practicable physical and process improvements that will ensure and enhance safe and reliable operation during the current PSR period which extends beyond the current licence period of Bruce A and Bruce B. The IIP will be updated regularly as a living document over the course of the PSR period as committed safety improvement initiatives and MCR outages are completed which will assure safe and reliable operation beyond the current licence period of Bruce A and Bruce B.

As indicated in [9] and [10], Bruce Power intends to submit a PSR-based licence application in Q3 2017, to include the IIP into the licensing basis consistent with Licence Conditions Handbook Section 15.2. The licence application will include a formal notice of intent for the first Bruce B unit (Unit 6) MCR outage program, a project execution plan and a return to service plan, in accordance with the Power Reactor Operating Licence (PROL) 18.00/2020 Licence Condition 15.2 and LCH Section 15.2. This transition to a PSR-based licence application process will also result in the IIP being updated, as required, to support the PSR process.

2. Background

Bruce Power and the Independent Electricity System Operator reached a commercial agreement in December 2015 to enable investments to extend the operational lives of Units 3-8 consistent with Ontario's Long Term Energy Plan including the replacement of major components. The agreement means Bruce Power will continue to invest in life extension activities for Units 3-8 in the current licence period, as well as future licence periods.

Bruce Power has developed an asset management strategy and plan in support of life extension of Units 1 to 8 as an essential part of their continued safe and reliable operation. This asset management strategy is being integrated with the REGDOC-2.6.3 Ageing Management [12] requirements and the Periodic Safety Review (PSR) process identified in Section 15.2 of the PROL [5][2] to ensure compliance with the current licensing basis and to assess the plant against modern codes and standards and where feasible to implement practicable safety improvements[3]. Figure 2 illustrates the overall framework which will be fully implemented at the end of the current licence period. The PSR process is implemented nominally every 10 years. This framework allows for updating of the IIP periodically through the PSR process. It should be noted that as the IIP is implemented, improvements realized are fed back to sustaining programs, PSR processes and Asset Life Management Options to sustain continuous improvement of plant safety and reliability.

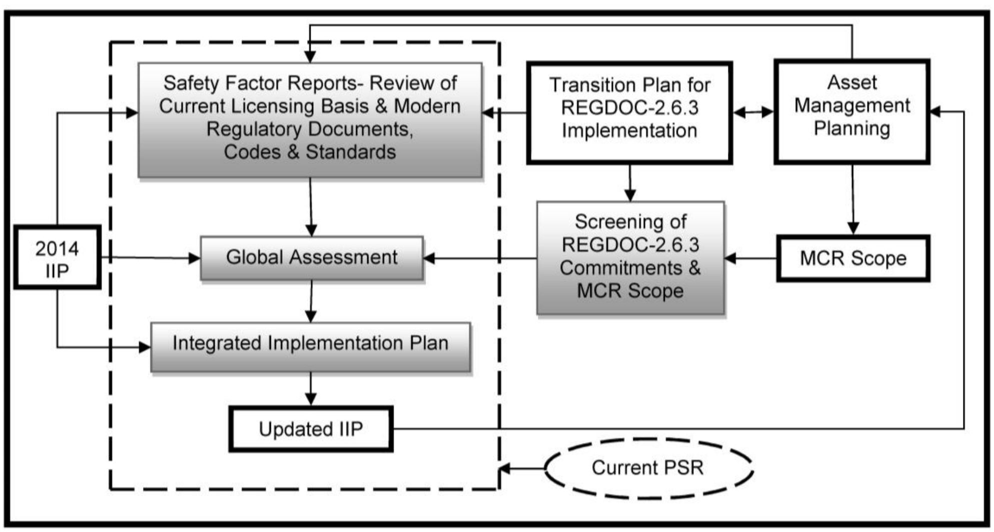

Figure 2: Bruce Power's Framework for Safe and Reliable Operation

[2] PROL 18.00/2020 [5] and LCH-BNGS-R000 [6] came into effect on June 1, 2015, and were used for the Bruce B PSR. However, PROL 15.00/2015 [7] and LCH-BNGSA-R8 [8] are the versions referred to in the Bruce A ISR, as these were in force when the assessments in the Bruce A SFRs were performed.

[3] The code effective date was August 31, 2014 for the Bruce A ISR and December 31, 2015 for the Bruce B PSR.

A Major Component Replacement (MCR) approach will be used to replace components such as pressure tubes, feeders and steam generators to extend the life of the units. Other asset management plan activities which cannot be implemented during planned maintenance outages will be also be executed during the MCR outage. The alignment of the MCR scope with asset management plans will support Bruce Power's goal for safe and reliable long term operation.

Conduct of this PSR supports the definition and timing of practicable and feasible opportunities for enhancing safety of Units 3 to 8 and allows for integration of MCR scope once timing is established and decisions are made. This review also includes the ongoing operation of Units 1 and 2, which have already been refurbished.

Bruce Power initiated the Integrated Safety Review (ISR) process for Bruce A in 2014 per the guidelines in CNSC Regulatory Document RD-360, Life Extension for Nuclear Power Plants 2008. In 2015, Bruce Power also initiated the PSR process for Bruce B in accordance with REGDOC-2.3.3. The Basis Documents for both activities [1] [2] have been accepted by the CNSC [3] [4].

Since Bruce A's ISR aligns with REGDOC-2.3.3 [13] and Bruce B's PSR complies with the regulatory document, terminology used in the PSR has been used for both Bruce A and B throughout this document.

The scope of the PSR, described in the ISR Basis Document [1] for Bruce A and PSR Basis Document [2] for Bruce B, is based on REGDOC-2.3.3, Periodic Safety Reviews [13][4] and IAEA Specific Safety Guide SSG-25, Periodic Safety Review for Nuclear Power Plants [15], and guidance from CANDU Owners Group Report, COG-10-9022, Principles and Guidelines for Undertaking an ISR for Nuclear Reactors in Canada, Revision 0, March 2011 [16] into consideration.

There are four (4) major phases of the PSR:

- PSR Basis Document, which defines the scope and methodology for the PSR is prepared and submitted first.
- PSR Basis Document is then used to conduct the safety reviews followed by preparation of the SFRs.
- Results of the SFRs are used as the input for GA, and the Global Assessment Report (GAR). The objective of the GA is to present the results of the PSR, both strengths and gaps, and to provide an overall assessment of the safety of plant. This is achieved via consolidation and integration of the findings of the SFR reviews into an overall assessment of safety, together with a list of on-going issue resolution activities and improvement opportunities. The GAR documents the overall conclusions, practicable corrective actions and other safety improvements to be considered.

[4] RD-360 [14] was superseded by CNSC REGDOC-2.3.3 [13] in April 2015, which was in draft at the time that the ISR Basis Document [1] was prepared. The draft version of CNSC REGDOC-2.3.3 stated that it was consistent with SSG-25, and the assessments in the Safety Factor Reports were performed on that basis. The issued version of CNSC REGDOC-2.3.3 also states that it is consistent with SSG-25, and therefore it is considered that the ISR envelops the guidelines in CNSC REGDOC-2.3.3.

- The results of the GAR are used to establish the corrective actions and safety improvements to be included in the IIP.

The results of the PSR process is documented and submitted to the CNSC, which includes:

- ISR and PSR Basis Documents [1] [2];
- Reports of each Safety Factor review [17] [18] [19] [20] [21] [22];
- GAR and IIP – This document integrates phases 3 and 4 to minimize duplication of information and for ease of finding relevant information and linkages between the GA and IIP.

3. Overview of the Global Assessment and the IIP Process

The GA and IIP process require four major inputs:

- PSR Basis Document;
- An assessment framework to conduct global assessment and develop the IIP, i.e., Global Assessment Methodology;
- Safety Factor Reports; and
- Planned initiatives and work in progress that is potentially relevant to GA and the IIP

As described in [1] [2], the guidance on generic global assessment and IIP development process in REGDOC-2.3.3 and SSG-25 has been tailored to proceed along the following steps:

- Develop an Assessment Framework;
- Integrate Review of Results from All SFRs;
- Consolidate Safety Factor Findings;
- Classify Safety Factor Findings and Develop GIs (Global Issues);
- Establish and integrate Improvement Initiatives outside PSR: IIP-2014, CNSC Action Items (AI) and Commitments, MCR Outage Improvements and other planned initiatives in place to enhance safe and reliable operation;
- Develop Global Improvement Opportunities (GIOs);
- Prioritize and Rank GIOs;
- Develop Corrective Actions (CAs);
- Prioritize and Rank CAs;
- Perform Risk Informed Decision Making (RIDM) (as needed);
- Develop the IIP and High Level CAs (as needed);

- Optimize the IIP;
- Prepare the IIP;
- Perform Global Assessment (Assessment of defence-in-depth, overall safety and justification for continued operation); and
- Prepare the GAR.

These steps form the basis for the GA and IIP development methodology. A description of each step is provided in Part I, Section 5 of this report. Figure 3 is a schematic representation and relationships of all the steps of the methodology leading to preparation of the GAR and IIP.

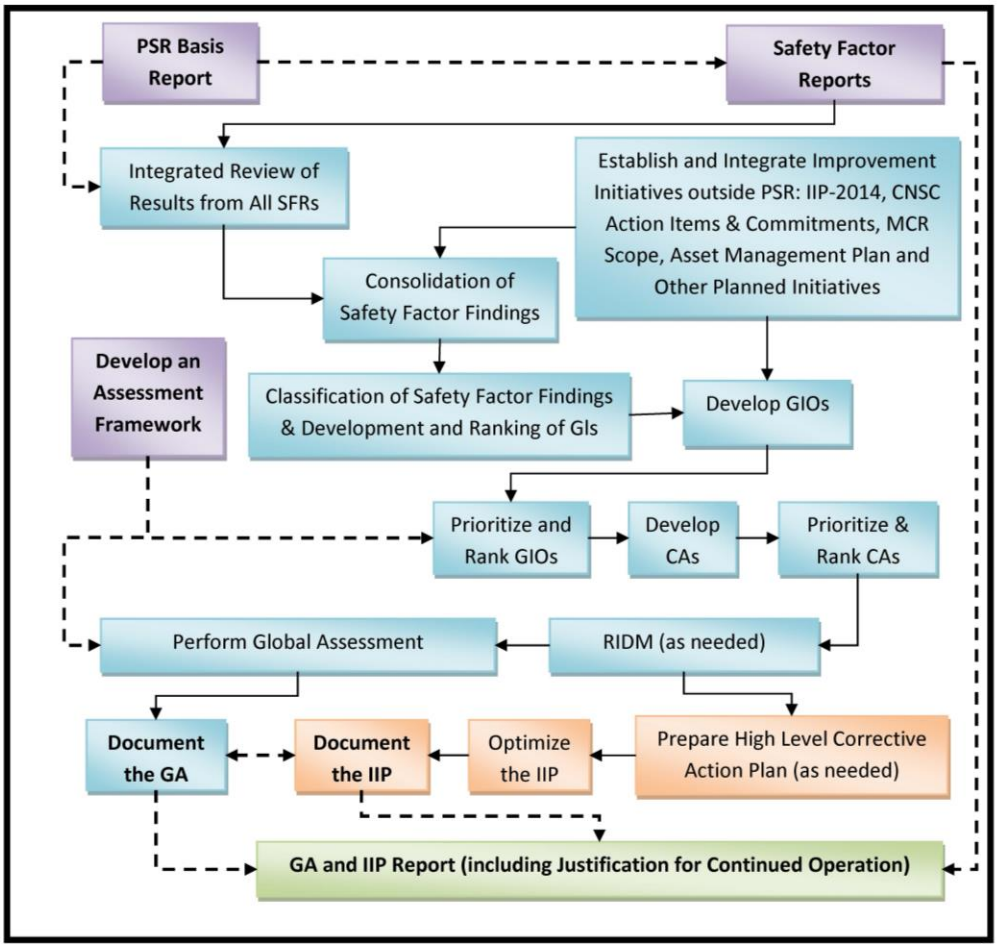

Figure 3: Global Assessment and IIP Process

4. Organization of the Report

The report is organized around the work done in each of the steps shown in Figure 3 and results obtained. The narrative represents the sequence of associated steps of the methodology to the extent possible as some activities have been performed in parallel.

Part I: Methodology

- Sections 1 to 3 describe the purpose, background and the overview of the Global Assessment and IIP process.
- Section 5 describes the steps of the overall methodology. The assessment framework is described in detail in Appendix C.

Part II: Integrated Review of Safety Factor Reports

- Section 6 summarizes all the strengths and gaps identified in the SFRs.
- Results are presented in tabular form in each section that follows together with a discussion and conclusion.

Part III: Global Assessment

- Section 7 summarizes the results of the review to address any overlaps, omissions, and interface issues of the findings from the SFRs and link all related micro-gaps where appropriate.
- Section 8 assesses consolidated negative findings (micro-gaps) from SFRs to determine if they should be considered as part of the IIP development and group them into GIs based on their topical similarities.
- Section 9 identifies initiatives based on input from Bruce Power. These are initiatives that have been identified through other processes outside the PSR that are related to the list of consolidated micro-gaps developed in Section 7, as well as other safety related improvement initiatives to be considered in the IIP.
- Section 10 consolidates micro-gaps from Section 8 and other initiatives from Section 9 together under entities known as GIOs that constitute the basis for the IIP scope.
- Section 11 presents results of ranking of GIOs in order of their priority to resolve them based on the magnitude and timeliness of the benefit to be achieved by their resolution.

Part IV: Integrated Implementation Plan

- Section 12 develops high level definition of CAs as CARDs (Corrective Action Requirements Definitions) for each GIO.
- Section 13 ranks all CARDs using the Global Assessment Framework (GAF), which provides for their prioritization for implementation.
- Section 14 summarizes the results of the RIDM assessments (if any) and those CARDs to be included or excluded from the IIP are identified.

- Section 15 summarizes the high level plan for CAs that defines scope and schedule. This plan comprises the initial IIP.
- Section 16 defines an optimal feasible sequence for implementing high priority corrective actions subject to the limitations imposed by scope, schedule, cost, outage length and frequency, resource availability and other constraints.
- Section 17 describes the IIP progress monitoring, change control and updates

Part V: Justification for Continued Operation

- Section 18 performs a defence-in-depth assessment of the plant based on the strengths, practicable safety improvements included in the IIP and the remaining micro-gaps (if any).
- Section 19 assesses the overall risk in terms of the overall safety goals set for continued plant operation
- Section 20 summarizes the justification for continued operation based on the results of all assessments presented in the report.

A simplified roadmap for the report is provided in Figure 4.

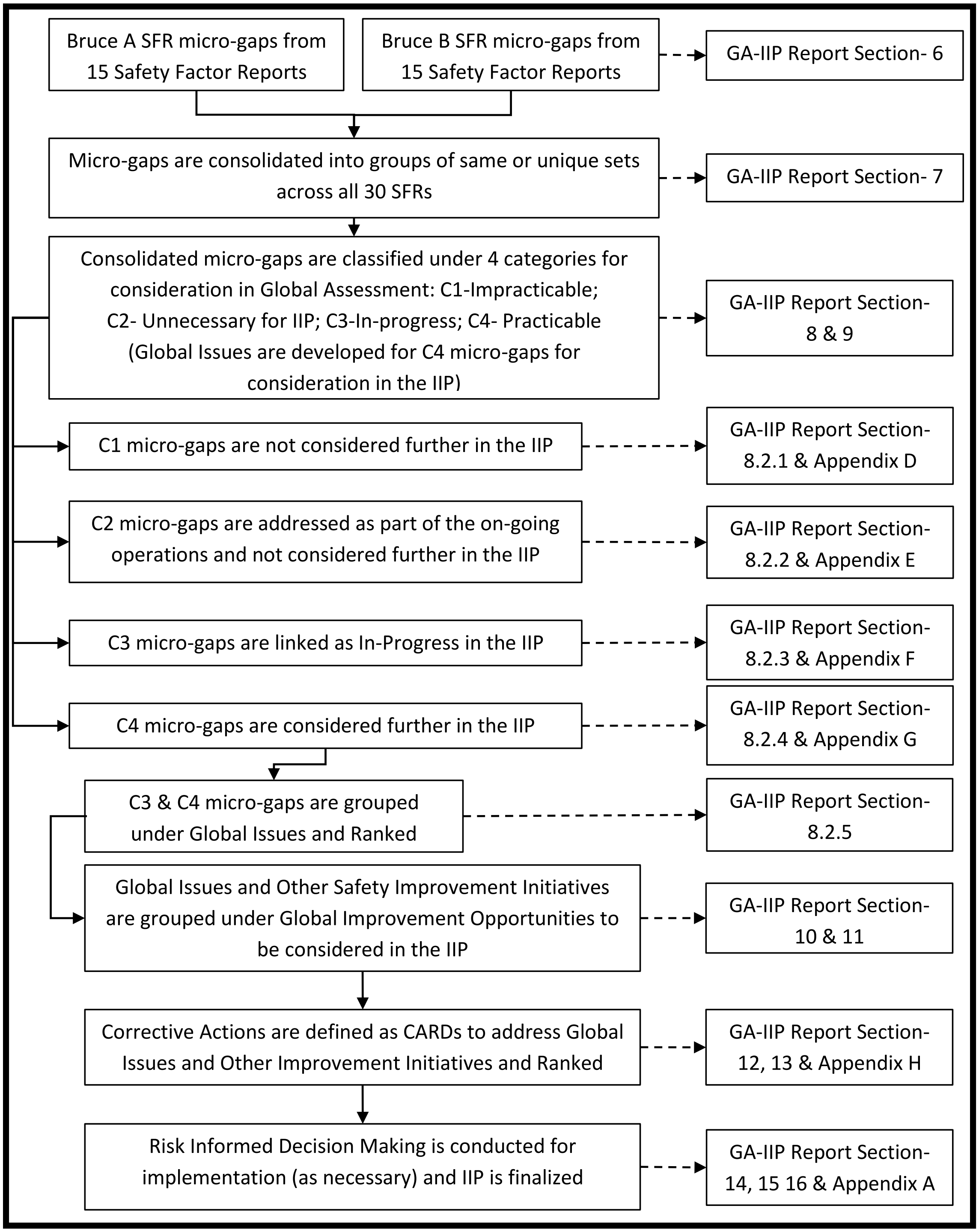

Figure 4: Simplified Roadmap for Global Assessment and IIP Report

5. Global Assessment Methodology

The GA methodology describes the conduct of assessments and the associated framework of the following elements:

- The Global Assessment Framework (GAF) is a common basis for systematically assessing the relative importance of addressing global issues in terms of aspects such as their safety significance and impact of their resolution. The same framework is used to assess the importance of practicable improvements and associated corrective actions. The GAF is presented in Appendix C.
- Integration of the results of the SFRs, in particular, the findings (both gaps and strengths) related to design and operation in terms of duplication (the same or similar micro-gaps identified in different regulatory documents, codes and standards or review tasks), and omissions (micro-gaps applicable to both plants identified only at one plant due to differences in the scope of regulatory documents, codes and standards used in each PSR), and interface issues;
- Assessment of interdependencies between consolidated micro-gaps, their assessment in terms of their practicability, necessity and status, consolidation into global issues and the safety significance of their aggregate effects;
- Definition of improvement opportunities and identification of relevant corrective actions already in place, or needed, for safety improvements to address individual and consolidated micro-gaps are developed;
- Integration of improvement opportunities resulting from the GA and on-going initiatives into an IIP under a set of GIOs and associated CAs;
- Assessment of the extent to which the safety requirements of defence-in-depth are fulfilled, taking into account improvements identified in the IIP;
- Qualitative evaluation of global risk associated with facility operation with any unresolved micro-gaps; and
- A final report (this report) summarizing the results of the SFRs, and documenting the GA and the associated IIP.

5.1. Integrated Review of Results from All SFRs

The main purpose of this first step of GA is to review and summarize the Safety Factor Report findings and identified micro-gaps and strengths of the current plant and its operation. This review also provides the opportunity to integrate all of the results summarized in Section 8 of each SFR, such that it is also possible to confirm the following:

- Which regulatory documents, codes and standards have been assessed in more than one Safety Factor that identified strengths/micro-gaps; and

- For each regulatory document, code or standard that was assessed in multiple Safety Factors, which clauses have unique and/or the same strengths/micro-gaps, i.e., duplication or overlaps.

This allows the comparison of strengths/micro-gaps associated with each clause/article of a regulatory document, code or standard to ensure consistency amongst SFR assessments, as well as further consolidation to be performed, described in Section 5.2.

It must be emphasized that micro−gaps have been identified against both requirements and guidance clauses in regulatory documents. The micro-gaps associated with 'Guidance' sections are considered to be NOT what the requirements ARE to be met, but rather HOW the preceding text that describes the requirements CAN be met. In this context, gaps related to guidance sections do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. The Global Assessment and IIP have been developed taking this principle into consideration.

In regards to guidance clauses, Bruce Power's official position on the statement that typically appears in the preface of regulatory documents that “should they [licensees] choose not to follow it [guidance], they should explain how their chosen alternate approach meets regulatory requirements.” is that if the licensee is required to meet guidance criteria, then these are requirements, rather than guidance. As such, Bruce Power does not agree with the gaps identified against guidance as requirements in regulatory documents."

For each SFR, results are reviewed and summarized under a standard set of headings in the GAR. If any modifications to the guidance provided in the PSR Basis Document have been made; they are also addressed together with the rationale for the change(s). The standard set of headings are:

- Objective
- Scope of the Review
- Regulatory Documents, Codes and Standards Assessed
- Overview of Applicable Bruce A and Bruce B Station Programs and Processes
- Interfaces with other Safety Factors
- Summary and Conclusions, including Table of Key Issues

5.2. Consolidation of Safety Factor Findings

The objective of consolidation of Safety Factor review findings is to:

- Address any duplication, omissions, and interface issues of the findings from the SFRs; and
- Link all related micro-gaps where appropriate.

The findings from each Safety Factor review, whether strengths or negative findings are based on the fairly narrow perspective of the Safety Factor. This step of global assessment provides for the consolidation of these findings to establish global findings through the removal of duplication and the broadening of context to make the findings comprehensive. This applies both to the strengths as well as the individual or collection of negative findings. Terminology used throughout this document with respect to negative findings is as follows:

- Individual negative findings are defined as "micro-gaps". Consolidation is performed at the micro-gap level.
- A collection of similar negative findings grouped and numbered in Section 8 Summary and Conclusions of each SFR is defined as a "macro-gap". Macro-gaps provide the link between the SFR and the GA-IIP, assuring traceability of each micro-gap to the source document.

Micro-gaps across all SFRs have been uploaded into the PSR database. Each micro-gap is provided with a unique database identification. The following additional information is included for each micro-gap:

- SFR Number
- Macro-gap Number and Title (as applicable)
- Reference to review task summary section and/or Appendix where the micro-gap is identified
- Regulatory Document, Code or Standard
  - Applicable section or clause
  - Text of the requirement relevant to the micro-gap
- Description of the micro-gap
- Type of the micro-gap (requirement, guidance, etc.)

5.2.1. SFR Micro-Gap Consolidation

Bruce A and Bruce B are two similar multi-unit stations, each with nearly identical units with a common design basis and each station operated within its own design and operating envelope. There is a common management system and governance covering both Bruce A and B. Hence, it is important to ensure that micro-gaps identified across all SFRs and both stations are consolidated in a manner to minimize duplication and eliminate omissions for their consideration in the GA and IIP development.

The purpose of this step is to consolidate individual micro-gaps across all Safety Factor Report findings by identifying common micro-gaps thereby eliminating duplication and identifying potential omissions of micro-gaps. In terms of consolidation:

- Duplication occurs as a result of micro-gaps that have been identified in different SFRs but which are the same or similar. The major reason for duplication is assessment of same or similar requirements or review tasks across different Safety Factors and using the same PSR process.

- Potential omissions of micro-gaps may occur as a result of the ISR being performed for Bruce A at an earlier time relative to the PSR performed for Bruce B. This may have resulted in differences in sets of regulatory documents, codes and standards and review tasks defined in the ISR/PSR basis documents, as well as the available information at the freeze dates identified in the basis documents for each station. Therefore, micro-gaps identified for one station may have been omitted for the other station, and any such omissions are reconciled in terms of their applicability to both stations as part of the GA process.

A “requirement based” comparison in terms of applicable regulatory documents, codes and standards and review tasks forms the basis for micro-gap consolidation. Bruce Power governance is “program/process based”, i.e., topical and as such in many cases the same requirement and associated micro-gap may appear for more than one governing document. Hence it may not always be feasible to consolidate micro-gaps based on a requirement but sometimes it may be more feasible to consolidate based on a program/process related topic.

Therefore the approach used is to consider two aspects, first from a requirement perspective then secondly from a program/process (topical) perspective.

Consolidation is implemented in three steps. Steps 1 and 2 address potential duplication. Step 3 addresses potential omissions.

Step 1: Requirement Based Consolidation:

Starting with the first requirement from a code or standard and review task where a micro-gap is identified, scrutinizing the remainder of the micro-gaps for coverage of the same or similar requirement(s) by using the table of key issues identified in Section 5.1, and summarized in Section 6 for each Safety Factor Report;

a. If similar micro-gap(s) or duplication is found:

    i. Linking these micro-gaps; and

    ii. Identifying which micro-gaps are duplicates or related across all affected Safety Factors.

    This step is repeated for each micro-gap.

b. If no similar micro-gap(s) or duplication is found, identifying remaining micro-gaps for potential consolidation in Step 2.

Step 2: Program/Process Based Consolidation:

Starting with the first remaining micro-gap, scrutinizing the remainder of the micro-gaps for coverage of the same topic or process;

a. If similar micro-gap(s) or duplication is found:

    i. Linking the affected micro-gaps; and

    ii. Identifying which micro-gaps are duplicates or related across all affected Safety Factors.

b. Identifying any micro-gaps that are unique.

At the end of this step micro-gaps that are unique are identified.

Step 3: Review of Station/Unit/SSC Specific Micro-gaps for Applicability to the Other Station/Unit/SSC:

Once the same or similar micro-gap consolidation is completed as described in steps 1 and 2, a review of remaining micro-gaps from step 2.b and consolidated micro-gaps from step 1 and 2.a is performed. The objective of the review of is to identify potential omissions in their applicability to other station/unit/SSC. Consolidated micro-gaps that are unique to one station/unit/SSC are reviewed for applicability to the other station/unit/SSC. Those micro-gaps that have been assessed as applicable to other station/unit/SSC are also included for consideration in GA and IIP development for the omitted station/unit/SSC. The micro-gaps identified based on the following are considered for applicability to eliminate potential omissions:

a. a finding based on a new code or standard or revised requirement included in a later revision of the same code or standard

b. a finding based on a review task, audit, self-assessment, FASA, etc. not evaluated in one station or due to a new or revised requirement in a review task or a new audit, self-assessment, FASA, etc.

If any omission(s) is found during this step, such micro-gap(s) is identified as applicable to the unit/station to address the omission(s). These micro-gaps are evaluated as applicable to both stations or units in the next steps of the GA and IIP development process.

The results of these reviews will be used in establishing the list of micro-gaps that are the same or topically similar to be used in development of GIs described in Section 5.5.

All three steps identified above are performed and documented in the PSR database.

5.3. Classification of Safety Factor Findings and Development of Global Issues

5.3.1. Introduction

The purpose of this step is to assess consolidated micro-gaps from the SFRs to determine their practicability, and if they are necessary safety improvements that should be considered as part of the IIP development. If they are deemed practicable and necessary, they are grouped into Global Issues based on their topical similarities. Input for this step is the results of the tasks described in Section 5.2.1.

Section 3.6 of REGDOC-2.3.3 states the following:

> *"To the extent practicable, the licensee shall resolve identified gaps with respect to applicable modern codes, standards and practices. The licensee shall use established processes to resolve identified gaps with the current licensing basis."*

The assessment for practicability is based on the detailed guidance provided in paragraphs 5.10 and 5.12 of SSG-25 [15] which states the following:

5.10. Negative findings should be divided into:

- *Deviations for which no reasonable and practicable improvements can be identified;*
- *Deviations for which identified improvements are not considered necessary;*
- *Deviations for which safety improvements are considered necessary.*

5.12. In the case of negative findings for which no reasonable and practicable improvements can be identified, the reason(s) should be documented and the issue revisited after an appropriate period of time to determine whether a practicable solution is available. For negative findings for which safety improvement are not considered necessary, the reason(s) should be documented and the action considered completed. Negative findings for which safety improvements are necessary, including updating/or extending of plant documentation or operating procedures, should be categorized and prioritized according to their safety significance. The categorization and prioritization of safety improvements may be performed on the basis of deterministic analyses, probabilistic safety assessment, engineering judgment, etc. Safety improvements from the safety factor reviews, together with safety improvements resulting from the global assessment, should be included in the operating organization's integrated implementation plan.

5.3.2. Assessment and Classification Scheme

Each micro-gap is assessed and then classified under one of the three groups based on SSG-25 [15] articles 5.10 and 5.12. One additional category includes those micro-gaps where there may already be an initiative to resolve it. For example, there may be an initiative in progress based on a transition plan to a new version of a regulatory document, code or standard specified in the PROL or included in the previous IIP or other commitments made to the CNSC. These categories are described in the following sub-sections. The process is represented in Figure 5.

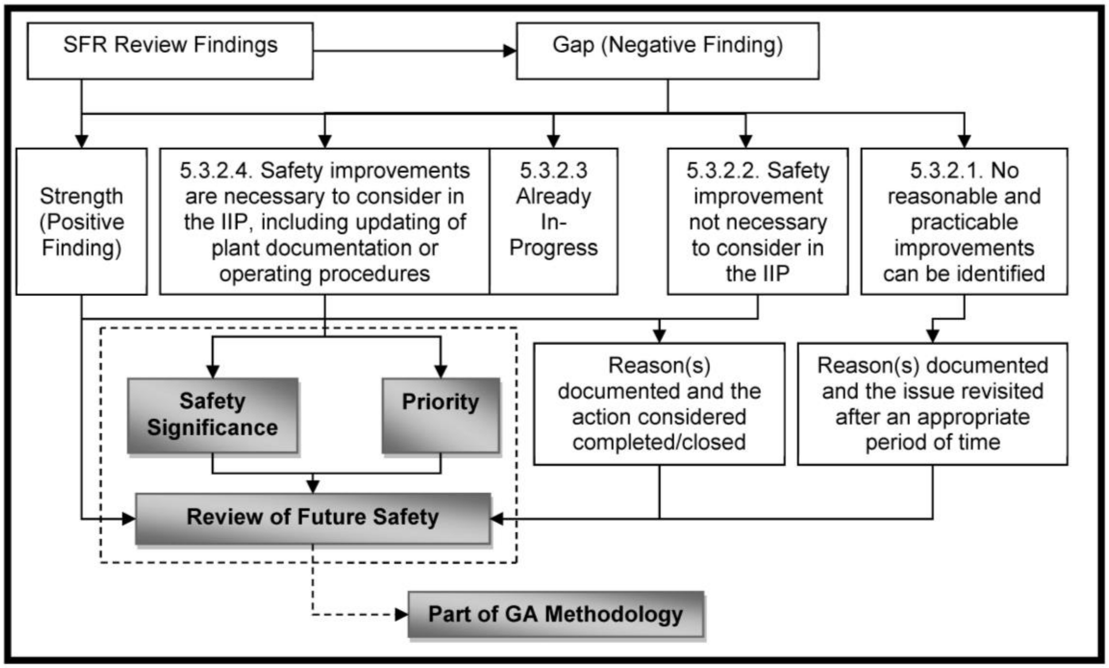

Figure 5: Assessment and Classification Process

5.3.2.1. Category 1: No Reasonable and Practicable Improvements can be Identified

Micro-gaps consolidated in Section 5.2.1 that are in this category could generally result from comparison against modern codes and standards and some international practices that have not been incorporated into the licensing basis of the plant as prescribed in the PROL. Some examples are:

- A generic requirement or principle that would require fundamental design changes to SSCs of the plant as a whole which cannot be accommodated within the current configuration of SSCs and plant layout. Due to the existing coupling of SSCs and their functional capabilities in the current design of the plant, changes to SSC(s) would also impact other physically connected or functionally related SSCs. Normally, compliance with this type of new requirement or principle can only be practically dealt with for a new plant as the physical and functional relationships have to be defined first as to meet the high level regulatory dose limits, safety goals, classification of SSCs and consequently design requirements. For example, the following principles and requirements would be considered in this category:
    - More conservative dose limits or safety goals, or higher safety margins or combination thereof than those currently in place due to new or updated requirements.

   - New requirements which were not explicitly considered in the original design of the plant, e.g., physical changes driven by evolving design philosophy for new NPPs as additional requirements or newer interpretation of principles such as redundancy, diversity, separation in terms of DID or improvement of safety goals.
   - Changes to the classification of SSCs or events or event sequences which lead to different or new design requirements as compared to the current design basis of SSCs.
- A practice that is not adopted by either the CNSC or the Industry in Canada for operating plants; e.g., requirements applicable to a new NPP or a different design technology such as a light water reactor (LWR).
- A requirement that is not adopted by either the CNSC or the Industry in Canada that fundamentally impacts the organization of the plant, its governance and processes which is not sustainable in terms of business objectives.

In many cases, the assessment of the current design demonstrates that there are other provisions in the design and operation that address the new requirement(s). Given the above considerations, and that both Bruce A and Bruce B Safety Reports demonstrate that the current licensing limits are met with adequate safety margins, individual micro-gaps that are classified in this category have a low safety significance, and would have high resource usage to realize the marginal benefits. Only when implemented collectively, these individual micro-gaps could result in meeting licensing limits applicable to new plants and deliver the expected safety benefit. However, implementing these micro-gaps collectively would require to build a new plant, which is beyond the objectives and scope of the PSR process.

For micro-gaps in this category, reason(s) for the classification are documented and the issue is revisited after an appropriate period of time (for example at the next PSR). New insights gained based on Operating Experience (OPEX), engineering and safety analyses performed since the last PSR on the plant response and capability against events and hazards not considered explicitly in the original design would provide additional input during revisiting such issues.

The integrated impact of not implementing these micro-gaps is addressed as part of Global Assessment.

5.3.2.2. Category 2: Safety Improvement Considered Unnecessary to Implement as Part of IIP

Those micro-gaps consolidated in Section 5.2.1 that do not have a significant impact on improving safety are considered as unnecessary to implement as part of the IIP. Micro-gaps in this category are considered to be practicable and would provide some benefit in improving the effectiveness of current processes in operation of the plant, or could provide some safety improvement through a modification of the plant design. Micro-gaps in this category are dealt with based on their source:

- Micro-gaps resulting from comparison against modern codes and standards and some international practices where there are alternative ways of addressing them within the current licensing framework and industry best practices. For micro-gaps in this category reason(s) for the classification are documented, and any follow-up actions or oversight are documented as appropriate, including associated ARs (Action Requests) such as an OPEX on international good practices, and the issue is categorized as “Closed” in the PSR database.

- Micro-gaps where safety improvements afforded by addressing them would be rendered unessential because the current DID provisions and the level of safety are sufficiently robust and their contribution to dose acceptance criteria and overall safety goals may be insignificant. For micro-gaps in this category reason(s) for the classification are documented and the issue is categorized as “Closed” in the PSR database.
- Individual micro-gaps resulting from less than adequate implementation of the current governance and associated procedures. These are mostly identified during the review of audit, FASA, peer reviews as part of review task assessments and in most cases specific corrective actions have already been identified for addressing them and are in progress. In this context, they do not present a generic process improvement opportunity that is safety significant and are dealt with through the current Corrective Action processes in place as appropriate.

In summary, for micro-gaps in this category reason(s) for the classification are documented and a list is provided to Bruce Power for their consideration to decide if any follow-up or additional oversight is required outside the PSR process. Such micro-gaps, based on Bruce Power’s input, are categorized as “Closed” in the PSR database and any follow-up actions for their implementation or oversight is documented including associated ARs (Action Requests).

Any micro-gaps for which closure actions are completed between the time of issue of the SFRs and the time of issue of the GAR will also be included in this category. Such micro-gaps, based on Bruce Power’s input, are categorized as “Closed” in the PSR database with the appropriate completion notes and references.

5.3.2.3. Category 3: Safety Improvement In-Progress

Micro-gaps consolidated in Section 5.2.1 that are the same as those that have already been identified in the previous PSRs or by other means are included in this category if there are initiatives or commitments in place to resolve them. The list of initiatives provided by Bruce Power includes, but is not limited to, those originating from the past assessments and ongoing activities appropriately cross-referenced to their original sources:

- MCR List of Initiatives;
- Capital Projects associated with Asset Management activities in support of safe long-term operation;
- CNSC Action Items;
- Transition Plans for compliance with new or updated regulatory documents, codes and standards;

- Fukushima Action Items (FAIs);
- CANDU Safety Issues (CSIs);
- Other Licence Submissions (including the latest IIP submitted to the CNSC); and
- Business plan and other initiatives that will improve safety.

Each micro-gap is checked against the list of initiatives listed above and the commitments and status of the corrective action(s) in place are investigated and documented. Each review results in one of the three sub-categories for such micro-gaps:

- If the associated corrective action(s) that will address the micro-gap(s) is completed, appropriate references pertaining to the completion are provided and the issue is considered as "Closed".
- If the associated corrective action(s) that will address the micro-gap(s) is in progress and being reported to the CNSC as part of the current IIP, appropriate references pertaining to the status are provided.
- If the associated corrective action(s) that will address the micro-gap(s) is in progress but is not part of the current IIP [11] [23], appropriate references pertaining to the status are provided and the issue is considered as "In-Progress". An example would be a micro-gap associated with a CNSC Action Item that is already in place, but not included in the current IIP. Such micro-gaps are linked to the applicable initiative(s) and included in the IIP directly.

5.3.2.4. Category 4: Safety Improvement Considered Necessary

This category includes the remaining micro-gaps consolidated in Section 5.2.1 that are not in Categories 1, 2 and 3 described above. Safety improvements associated with these micro-gaps are classified to be necessary for consideration in the development of the IIP. Generally these include:

- Maintenance, repair or replacement of plant SSCs important to safety and reliability;
- Engineering assessments and analyses supporting continued operation for the assessment period;
- Practicable design modifications and improvements to the current structures and equipment to ensure compliance with the current design basis and expectations in the modern codes and standards; and/or
- Updating or extending of plant documentation or operating procedures.

5.3.2.5. Documentation of Results

At the end of this step, all micro-gaps consolidated in step 5.2 are classified under the four categories described:

- Category 1 includes micro-gaps impracticable to implement and generally with low safety significance and safety benefit;
- Category 2 includes micro-gaps practicable to implement through the established processes to resolve or unessential to implement;
- Category 3 includes practicable micro-gaps that are already being implemented as part of the current IIP;
- Category 4 includes practicable micro-gaps with some safety benefit to consider for implementation in the IIP.

In summary, in addition to Category 3 micro-gaps which are already in progress as part of the current IIP, a set of Category 4 micro-gaps based on the Safety Factor review findings that should be considered in the development and update of the IIP are identified. Those micro-gaps classified as Category 1 and Category 2 are also considered in the overall review of future safety as part of the GA process.

5.3.3. Development and Ranking of Global Issues

The purpose of this step is to develop and rank Global Issues (GIs) making use of the insights gained from the tasks described in Section 5.2.1. Category 4 consolidated micro-gaps identified as a result of the tasks described in Section 5.3.2.4 are reviewed against each other for common features thereby grouping them as a smaller set of GIs. The approach used is to review two aspects of Category 4 consolidated micro-gaps based on Section 5.3.2.4 results; first from common or similar requirement(s) perspective and if no common or similar micro-gap(s) are identified then secondly from a common process perspective. Similar requirements may constitute a set of consolidated micro-gaps arising from a general requirement in a modern standard such as acceptance criteria for structural integrity or classification of postulated initiating events for safety analysis. Common process consolidated micro-gaps may constitute a set of consolidated micro-gaps arising from a review task and a modern standard. For example, a micro-gap arising from a modern standard in training may be consolidated with an improvement in a training gap arising from a review task covered under updating governance associated with training. The approach is implemented in two steps.

Step 1: Requirement Based Grouping

1. Starting with the first requirement where an SFR micro-gap, or consolidated micro-gaps, is identified scrutinizing the remainder of the micro-gaps for coverage of the same requirement by using the results of the tasks described in Section 5.2.1;
2. If a similar set of consolidated micro-gap(s) associated with a requirement is found:
   a. Linking, in the PSR database, the affected micro-gaps to a single GI; and
   b. Providing a clear GI title.
3. If no similar micro-gap(s) is found, identifying remaining micro-gaps to be considered for step 2 of the process.

Step 2: Process Based Grouping

1. Starting with the first remaining SFR micro-gap, or consolidated micro-gaps, scrutinizing the remainder of the micro-gaps for coverage of the same issue by using the results of Section 5.2;
    a. If commonality is found:
        i. Linking, in the PSR database, the affected micro-gaps to a single GI including (if any) duplicates for completeness; and
        ii. Providing a clear GI title.
    b. If no commonality is found:
        i. Creating a GI that for the individual micro-gap; and
        ii. Providing a clear GI title.

At the end of this step all Category 4 micro-gaps identified per Section 5.3.2.4 are mapped to a GI.

Ranking of GIs is performed per the Global Assessment Framework (GAF) described in Appendix C. Each GI is ranked at Tier 2 of the Value Tree as described in Appendix C. The objective of developing an assessment framework is to devise a systematic methodology and establish a common basis for assessing the relative importance of addressing Global Issues in terms of aspects such as their safety significance. The same framework is also used to assess the importance of practicable improvements and associated corrective actions for the development of the IIP.

More specifically, the process allows for importance ranking and prioritization of the issues and potential improvements identified through the PSR and other assessment activities. This is achieved through a multi-objective, multi-attribute decision support model formulated as follows:

- The multi-objective nature of the problem is described by decomposing overarching objectives into a hierarchical structure of sub-objectives called a value tree. The often conflicting nature of sub-objectives is accommodated through the allocation of relative weights to objectives attached to the same branch level of the value tree. Higher weights are assigned to branches for which enhancements provide the greatest benefit to safety, thereby risk-informing the value tree;
- A scoring system is devised that allows the decision maker to express preferences for resolving issues on a 5-point scale for each of two attributes: impact and time-to-take-effect. The impact score will take into account aspects such as contribution to defence-in-depth and safety significance, particularly impact on achieving safety goals;
- The impact and time scores are combined to produce an overall utility score for each issue that reflects a preference for resolutions that achieve high impact in a short time, but weigh impact somewhat higher in importance than time. Higher scores denote a greater preference for resolution, again risk-informing the process by placing priority on issue resolution that will have the greatest value in supporting the underlying objective; and

- Finally, the value ranking of resolving an issue is calculated as the product of the relative weight of the corresponding objective and the utility score of the issue.

The resulting prioritization and ranking framework is embedded in the PSR database. The value tree has three tiers below the cardinal objective. The first two tiers are utilized in the development, ranking and prioritization of Global Issues. The third tier is utilized in the development, ranking and prioritization of corrective actions to address Global Issues

Development and ranking of GIs is performed in the PSR database. A specific verification shall be performed to ensure that all Category 3 and 4 micro-gaps are linked to a GI.

5.4. Establish Improvement Initiatives Outside of PSR: MCR Outage Scope, Asset Management Activities, IIP-2014 and Bruce Power Improvement Projects and Initiatives

The purpose of this step is to collect and integrate all non-SFR initiatives that have been identified through other assessments or initiatives outside the PSR, as well as any other improvement initiatives to be considered in the IIP based on input from Bruce Power.

The list of initiatives provided by Bruce Power will include, but not be limited to, those originating from the past assessments and ongoing activities appropriately cross-referenced to their original sources. The same list of initiatives listed in Section 5.3.2.3 is used in this step.

As part of Section 5.3.2.3, the list of initiatives were reviewed against each micro-gap to establish if there are any initiatives when completed will support resolution of the issue associated with the micro-gap.

Those improvement initiatives that will improve safety that are not related to any of the micro-gaps as part of Section 5.3.2.3, but will support and enhance safe and reliable operation during the PSR period and beyond will be listed as initiatives to be considered during the GA and in the development of IIP. The safety improvements related to MCR and Asset Management initiatives will be selected in accordance with the screening process described in Section 5.4.1.

The result of this step is a list of consolidated safety-related improvement initiatives (if any) as part of:

- IIP initiatives that are in progress (in this case the 2014 IIP [11]);
- Additional MCR and Asset Management initiatives to be included in the IIP based on the screening process described in Section 5.4.1; and
- Other initiatives planned based on input from Bruce Power.

The approach used in step 5.4.1 results in the integration of:

- The latest IIP submitted to the CNSC with this PSR;
- Allows for augmentation of the list of safety-related improvement initiatives to be considered for GA and IIP and its periodic update on a continuous basis; and

- Bruce Power's long-term plans and commitments for safe and reliable operation of Bruce A and Bruce B beyond the current PSR or PROL.

This approach is illustrated in Figure 2. It should be noted that the Asset Management Planning, Transition Plan for REGDOC-2.6.3 implementation [23] and MCR scope related boxes in Figure 6 are specific to this PSR. They have been explicitly identified as they have a direct impact on the current PSR and licensing process.

5.4.1. Guidance for Screening MCR and Asset Management Scope for Inclusion in the Global Assessment and IIP Development

This section summarizes the guidance on how to screen activities supporting safe long term operation from Bruce Power's MCR and Asset Management programs for consideration in the Global Assessment and Integrated Implementation Plan for Bruce A Units 0, 3-4 and Bruce B Units 0, 5-8 [24].

As stated in Section 2 of REGDOC-2.3.3 [13]:

The objectives of a PSR are to determine:

1. *the extent to which the facility conforms to modern codes, standards and practices*
2. *the extent to which the licensing basis remains valid for the next licensing period*
3. *the adequacy and effectiveness of the programs and the structures, systems and components (SSCs) in place to ensure plant safety until the next PSR or, where appropriate, until the end of commercial operation*
4. *the improvements to be implemented to resolve any gaps identified in the review and timelines for their implementation*

MCR outage scope includes a wide variety of activities that extend beyond those required by the provisions of Section 15.2 Continued Operations of the PROL. From the MCR outage scope, only activities essential to meet objectives 2 and 3 of the PSR (as stated in Section 2 of REGDOC-2.3.3) need to be included in the IIP.

5.4.1.1. Screening of Asset Management Inputs of Major Component Replacement Scope

The MCR outage schedule and scope are dictated by life extension activities driven by Bruce Power's asset management processes, where major component replacement is planned above and beyond what may be expected in a normal planned maintenance outage.

These activities include the replacement of fuel channel assemblies, feeder piping and steam generators, etc., which can only be performed in an MCR outage in order to extend the unit life up to another 30 years.

Although not explicitly shown in Figure 2, additional IIP initiatives may also be identified and implemented during an MCR outage of a unit undergoing refurbishment as part of the PSR process.

Safety improvements that can only be made during an MCR outage – such as regulatory inspections that cannot be performed under planned outage conditions – are by default considered during the GA for inclusion in the IIP to meet objectives 2 and 3 of the PSR. Therefore, they are not discussed further in this report.

In summary, within the 10-year PSR window, the MCR scope of the IIP is limited to a unique subset of asset management activities that must satisfy both of the following conditions:

a. The activity cannot be implemented as part of on-going plant operations, i.e., during power operation or planned outages; and

b. Completion of the activity assures the licensing basis will remain valid following the MCR outage for continued safe operation within or beyond the current PSR interval.

Bruce Power's MCR outage scope has two parts:

1. Replacement of major components such as fuel channels, feeders and steam generators. These ***must*** be included in the IIP because they satisfy both conditions a and b.

2. Balance of Plant. In the context of MCR outage scope, this means any work not covered under the MCR program in 1. above. Balance of Plant scope is driven by three considerations:

   a. Essential asset life management work based on asset life limitations described in the Life Cycle Management Plans where:

      (i) Components must be replaced within the MCR window based on end-of-life limits; and

      (ii) Scope, logistics and duration of work is such that it cannot be executed effectively on-line or during a planned outage.

   Bruce Power Engineering is responsible for allocating all Asset Life Management scope work to on-line work or the appropriate outage windows, including the MCR outage. Asset Life Management scope in this category ***must*** also be included in the IIP because it satisfies conditions a and b.

   b. Planned work that falls within the MCR outage window. This includes all scheduled maintenance and replacement work not covered in (2a) that happens to be within the MCR window. For example, cable replacement is an ongoing activity and, as such, all cables that were scheduled during the time period of the MCR outage will be replaced within the MCR window. In this case, since the work is not driven by component end-of-life considerations related to condition a or duration/access afforded by the MCR window as related to condition b, it does ***not*** need to be included in the IIP.

   c. Capital projects essential for sustaining current plant operations that fall within the MCR window. For example, if this work is not driven by component end-of-life considerations related to condition a(i), or duration/access afforded by the MCR window as related to condition a(ii), then it does ***not*** need to be included in the IIP. It should be noted that capital projects are reviewed separately as part of the Global Assessment to determine if they will help resolve micro-gaps identified in the Safety Factors Reports. If so, these capital projects ***are*** included in the IIP in support of the safety improvement initiative.

5.4.1.2. Screening of REGDOC-2.6.3 Transition Plan Commitments in Support of PSR

Bruce Power submitted a transition plan to the CNSC for REGDOC-2.6.3 implementation in support of the licence renewal process for the current licence [25]. This transition plan mainly focuses on a review of the current programs and procedures in support of aging management against the requirements of REGDOC-2.6.3 and the appropriate steps to address any potential gaps to ensure compliance. Implementation of the transition plan for REGDOC-2.6.3 may include program/procedure improvements, augmentation of safety assessments and follow-up actions. Some of these follow-up actions may need to be implemented beyond May 31, 2020, when Bruce Power's current operating licence ends and up to the start of the MCR outage of each unit to maintain the licensing basis of the associated units. The MCR outage start dates for Units 3 to 6 are within the current 10-year PSR window.

Implementation requirements for REGDOC-2.6.3 are stated in Section 6.1 of the LCH [6]. Revision R002 of the LCH (effective February 1, 2017) states:

> *"Bruce Power has submitted the transition plans to meet REGDOC-2.6.3 on December 12, 2014.*
>
> *The Bruce Power integrated aging management program is in compliance with REGDOC-2.6.3 with the exception of the LCMPs. Final implementation of all LCMPs, with the exception of feeders, fuel channels and steam generators, has been completed. The REGDOC-2.6.3 compliant LCMPs for feeders and the steam generators are expected to be issued by April 2017 and the LCMP for the fuel channels is expected to be issued by June 2017."*

Furthermore, a description of the overall Asset Management process is provided in References [26], [27], and [28].

There are safety assessments and follow-up actions, many of which are currently ongoing, that relate to confirmation of adequate safety analysis and operating margins based on the integrated effects of aging[5] and required to maintain the licensing basis of the SSCs up to the start of MCR of the affected units. As a result, they will be included in the IIP.

5.4.1.3. Screening of MCR and REGDOC-2.6.3 Transition Plan Commitments for Inclusion in IIP

A simplified representation of the process for screening of major component replacement and asset management safety integration for inclusion in the IIP is shown in Figure 6. Any initiative from either source can be screened by simply asking one question for each type of initiative.

[5] Integrated effects of ageing on the safety case includes assessment of combined effects of different ageing mechanisms of affected SSCs taking into consideration their current and projected conditions and the impact of any proposed plant design, operational or configuration modifications to offset aging impacts

- **For MCR initiatives**: Does the initiative satisfy the following conditions?
  a. The activity cannot be implemented as part of on-going plant operations, i.e., during power operation or planned outages; and
  b. Completion of the activity assures the licensing basis will remain valid following the MCR outage for continued safe operation within or beyond the current 10-year PSR interval.
- **For REGDOC-2.6.3 transition plan commitments**: Do program/procedure, safety assessments and improvements relate to confirmation of adequate safety analysis and operating margins based on the integrated effects of aging and required to maintain the licensing basis up to the start of MCR and after MCR?

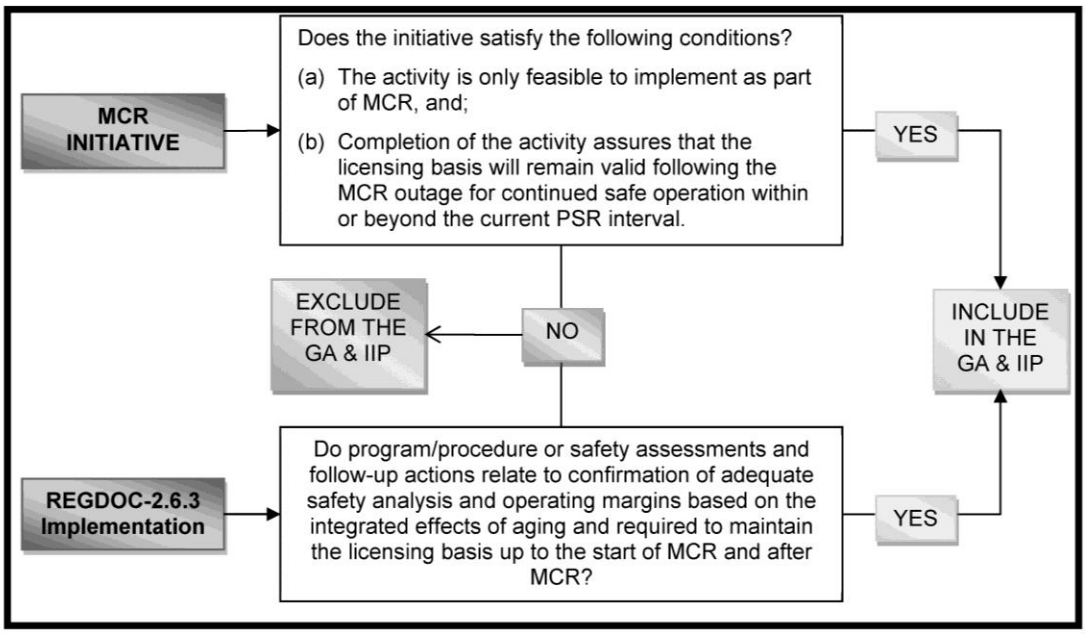

Figure 6: Simplified Representation for Screening MCR and REGDOC-2.6.3 Implementation for Inclusion in IIP

5.5. Development of Global Improvement Opportunities

Global Issues (GIs) developed from the activities described in Section 5.3 and other safety related improvement initiatives identified in Section 5.4 are integrated under entities known as GIOs. These new groupings have been defined as GIOs, because some of the micro-gaps in the GIs may be related to currently planned initiatives based on Section 5.4, which means that other improvement opportunities are planned independent of this PSR as part of the ongoing plant operation and licensing requirements. The approach used is the same as that used for development of GIs in Section 5.3.3. This step may involve expansion of the GI list due to integration of relevant safety related improvement initiatives identified in Section 5.4. Each safety related improvement initiative identified in Section 5.4 is reviewed against those identified in Section 5.3 to establish commonalities and mapped to the appropriate GI. After this mapping, each GI is defined as a GIO integrating safety related improvement initiatives from Sections 5.3 and 5.4. If an initiative from Section 5.4 cannot be integrated with an existing GI from Section 5.3, a new GIO is created.

One important aspect of GIO definition is that all micro-gaps and initiatives consolidated under each GIO must belong to the same sub-objective (Tier 2) of the value tree described in Appendix C so that their relative ranking and prioritization can be performed in a consistent manner.

In terms of their content in many cases GIOs will be the same as GIs for those GIs addressing SFR micro-gaps only. Some GIOs may not contain any SFR micro-gaps, for example, some of those associated with MCR scope. In summary, GIOs will contain an integrated set of PSR based improvement opportunities from SFR micro-gaps and those planned initiatives identified from processes other than PSR. In the PSR database, GIs developed per Section 5.3 have the same designation as GIOs.

5.6. Prioritization and Ranking of Global Improvement Opportunities

The purpose of this step is to arrive at a list of GIOs ranked in order of priority based on the magnitude and timeliness of the benefit to be achieved by solving them. Note that this ranking only indicates the importance of the GIO, but not the feasibility of the associated corrective actions subject to constraints of cost and time or other intangible considerations. The latter is considered as part of development of the IIP. The ranking and prioritization step entails the following:

- Use the GAF described in Appendix C, as implemented in the PSR database, to assign each GIO to a second tier objective in the value tree. In so doing, the GIO assumes the same priority as the Tier 2 objective as expressed in the weight of the objective;
- Taking into consideration the nature of potential corrective actions for the GIO use the GAF to evaluate the impact and time-to-take-effect of resolving the GIO. In so doing, a two parameter utility score is assigned to the GIO;
- Calculate a ranking number for the GIO by multiplying the assigned weight and score; and
- Arrange the GIOs based on ranking number from highest to lowest to arrive at a ranked list.

5.7. Develop Corrective Actions

This step provides for the identification and high level definition of CAs to address each of the GIOs.

The development of the CAs adheres to the following principles:

- An integrated approach to remove scope overlaps and optimize available time and resources – corrective actions identified either through the SFRs or other sources are integrated and consolidated;
- Deterministic and probabilistic safety assessment insights (e.g., where applicable, contribution to Core Damage Frequency (CDF) or safety goals or reduction in public dose, etc.) are utilized to the extent practicable in establishing risk importance, prioritization, and ranking of improvement opportunities that will be subject to a RIDM process;
- Contribution of corrective actions to defence-in-depth and the fundamental safety functions are taken into consideration;
- Corrective actions to be taken are evaluated in terms of their contribution to actual benefit to safety taking into consideration how soon it will be effective once implemented;
- Alternative means of achieving the safety benefit are considered if adequate interim measures can be implemented that are commensurate with the safety significance. The safety impact of not implementing a particular improvement is also considered as one of the options in all cases;
- Interface with Bruce Power stakeholders during the development of the CAs.

Once Category 4 micro-gaps and initiatives mapped from other sources (Section 5.4) are consolidated under a CA, they are linked in the PSR database. This ensures traceability of all micro-gaps mapped from the SFRs (Section 5.3.2.4) and initiatives mapped from other sources (Section 5.4). CAs are designated as CARD (Corrective Action Requirements Definition) with a serial number in the PSR database.

One important aspect of the CA definition is that all micro-gaps and initiatives consolidated under each CA must be chosen such that they can be mapped under a single Tier 3 sub-objective of the value tree described in Appendix C so that their relative ranking and prioritization can be performed in a consistent manner.

5.8. Prioritization and Ranking of Corrective Actions

The prioritization and ranking of CAs uses the Global Assessment Framework (GAF) and follows the same process as that of GIOs, the only difference being that CAs are assessed against the third tier of the value tree. Using the GAF described in Appendix C, as implemented in the PSR database, the ranking and prioritization step entails the following:

1. Associate each CA with the second tier objective in the value tree that corresponds to the branch associated with the highest ranked GIO it is intended to address;

2. Assign the CA to the appropriate third tier sub-objective that the CA will support under the same second tier branch. In so doing, the CA assumes the same priority as the sub-objective as expressed in the weight of the sub-objective;
3. Taking into consideration the nature of the CA use the GAF to evaluate the impact and time-to-take-effect of resolving the GIO. In so doing, a two parameter utility score is assigned to the CA;
4. Calculate final score for the CA by multiplying the assigned weight from Step 2 and utility score from Step 3; and
5. Arrange the CAs based on final scores obtained in Step 4 from highest to lowest to arrive at a ranked list.

5.9. Perform Risk Informed Decision Making (as needed)

The need to perform a RIDM assessment is determined on a case-by-case basis, based on the scope, schedule and cost considerations associated with each CA and the expected safety benefit from its implementation. For example, a RIDM assessment would be required in cases where:

- The associated costs are so extensive that implementation of similar or higher ranked CAs may be delayed; or
- Other considerations such as Bruce Power's asset management plan expectations.

RIDM is performed in accordance with B-REP-03611-00004 Risk Informed Decision Making Process. The results of each RIDM assessment will be included in the GAR as an Appendix.

The output of this step is a final list of practicable Corrective Actions that serve as input to the IIP.

5.10. Integrated Implementation Plan

5.10.1. Purpose

The objective of integrated implementation planning is to arrive at a single comprehensive set of cost-effective improvement initiatives that eliminates duplication of effort and provides for maximum synergy by:

- Documenting planning for all of the corrective actions and safety improvements that will be implemented based on their relative ranking in terms of their utility based on their safety significance and time to become effective; and
- Specifying the schedule for implementing the resulting corrective actions and safety improvements.

5.10.2. Scope

The IIP includes the following activities and processes:

- The process and PSR database tool, which demonstrates traceability by providing appropriate references to the GAR;
- The processes used for determining the scope, including prioritization and scheduling of corrective actions and safety improvements;
- The process and the methodology used to ensure that corrective actions and improvements that have the greatest impact on safety and reliability are prioritized so that they can be implemented in a timely manner;
- Processes to be used for identification and management of project risks and controls;
- Processes to be used to track the progress and completion of the corrective actions and safety improvements; and
- The basic principles for the change control process to update the planning in the IIP.

The CAs identified during Global Assessment may be new or may involve previously identified or ongoing activities, such as those included in the IIP for 2014 [11].

5.10.3. Integrated Implementation Plan Methodology

The development of the IIP entails the following steps:

1. Develop a High Level Corrective Action Plan for each Corrective Action;
2. Optimize the IIP; and
3. Document the IIP.

The development of the IIP adheres to the following principles:

- Ranked and prioritized corrective actions are further integrated to optimize available resources and time and to maximize the safety benefit; and
- Unit or station specific initiatives are specified accordingly.

It is noted that ranking and prioritization as described in Section 5.8 is still valid for the High Level CAs and hence need not be repeated as long as the corrective actions associated with each CA are not integrated with another CA for optimization or changed. In such cases ranking and prioritization of the integrated CAs will be performed as described in Section 5.8.

Each of these steps is discussed in more detail below.

5.10.3.1. Develop a High Level Plan for Corrective Actions

The input for this step is a set of prioritized and ranked CAs from activities described in Section 5.8. Preparation of the plan includes:

- Definition of a high level scope; and
- Definition of a high level schedule.

In order to minimize potential duplication and the effort associated with preparing CAPs, Project Plans or Action Tracking actions or similar documentation that are already in place must be used as the basis for establishing the need for developing the CAPs. Such documents can be used as the CAP when deemed appropriate. In this context, CAPs will be prepared on an as required basis.

Some CAs may be defined in greater detail than others depending on their implementation schedule and associated prerequisites. For example, on-going activities associated with some of the CAs will be well defined with a detailed scope, schedule and execution plan. While other CAs may be in their project initiation phase and hence less defined. Hence, the level of detail only reflects the stage at which the CAs are at with respect to implementation. Periodic updates on the progress of the IIP implementation will provide further details on all CAs commensurate with their committed target completion dates.

5.10.3.1.1. High Level Scope

A high level scope for each CA associated with a GIO is based on input from Bruce Power which integrates the improvements identified and related projects, planned actions to close the related CNSC Action Items (AIs), planned maintenance, inspections, and any other activities. Appropriate links to the relevant Project Plans, Bruce Power Action Tracking System ARs (Action Requests), Regulatory commitments, etc., will also be identified as part of this step.

The high level scope will identify:

- Objective(s);
- An integrated set of corrective action(s) to meet the set objective(s) – both new and those that are in progress;
- Details of the initiatives and associated issues being addressed by the corrective action including associated references (e.g., CNSC AI);
- An assessment of the applicability of the corrective action across Bruce A and Bruce B units;
- A description for each corrective action;
- References to project plans or action tracking actions; and
- Any long lead aspects in the planning of corrective actions.

5.10.3.1.2. High Level Schedule

The high level schedule includes:

- List of corrective actions
- Where applicable, a sequence of corrective actions and the prerequisites of the work needed to be performed to complete the corrective action; and
- Target completion dates for each corrective action.

In order to minimize potential duplication and the effort associated with preparing high level implementation plans, Action Tracking ARs, outage or project plans and similar documentation that are already in place must be used. Such documents can be used as the high level implementation plan when deemed appropriate. As such, any schedule already developed or planned for development will be incorporated in the IIP based on input from Bruce Power.

In summary, the level of detail with respect to scope and schedule will be summarized in the updates to the IIP and will be commensurate with the current status, timeframe and prerequisites for completion of the activities required to implement the CA.

5.10.3.2. Optimize the Integrated Implementation Plan

The purpose of this step is to determine the optimal feasible sequence for implementing high priority corrective actions subject to the limitations imposed by scope, schedule, cost, outage length and frequency, resource availability and other constraints. An important consideration of this step is to review the relationships between corrective actions irrespective of their ranking and based on implementation effectiveness. Those corrective actions or their elements which may be a pre-requisite to another or those where their implementation and timing present economies of scale would be planned accordingly.

Specifically, an integrated review with the MCR plans and other asset management initiatives and associated corrective actions will be performed periodically to remove potential duplication, identify opportunities for optimization of scope, resource needs and schedule.

5.10.3.3. Document the Integrated Implementation Plan

The results of the steps outlined above are documented to include the following:

- An IIP in the form of proposed list of safety improvements, including their safety significance, prioritization and timing for implementation.
- The IIP is listed according to the CNSC's safety and control areas so as to facilitate the CNSC's review. Appendix C, Table 38 shows the relationship between the CNSC Safety Control Areas and PSR Safety Factors and the Value Tree Tier 2 Objectives
- To ensure the success of the IIP, the following elements will be in place:
    - Organizational arrangements in place to execute the IIP;

- Governance applicable to the delivery of the IIP;
- Where necessary, scope, schedules and dependencies, for the earlier tasks that have an impact on critical path;
- A high level definition of resources and a resourcing plan if constraints are specified with respect to availability of resources;
- The mechanism for overall integration, peer or independent review and oversight; and
- Reference to a procedure that will govern change control of the IIP, or change control principles that will subsequently be incorporated into an IIP change control procedure.

5.11. Perform Global Assessment

The objective of the GA is to present an overall evaluation of the safety of Bruce A and Bruce B, taking into account a balanced assessment of all findings identified in the PSR, including the improvements in the IIP.

The assessment includes:

- the SFR findings on the compliance of the plant design, SSC condition and operation with the PROL,
- the set of strengths and global issues resulting from the consolidation of Safety Factor findings, and
- practicable Corrective Actions identified in the IIP are taken into account in the Global Assessment.

The GA involves the formulation of arguments that seek to justify a position that it will be safe to continue operating Bruce A and Bruce B for the PSR period and beyond. This formulation will therefore address the following:

1. A global assessment based on the aggregate effect of the findings resulting from all SFRs, taking the proposed corrective actions and safety improvements into account, together with their relative importance as expressed by their ranking numbers.
   a. An assessment of defence-in-depth taking into consideration the current plant and its operation and contribution of initiatives included in the IIP and strengths identified in Safety Factor reviews.
   b. A qualitative assessment of overall risk in terms of deterministic dose acceptance limits in the PROL and probabilistic safety assessment of Bruce Power’s safety goals including those micro-gaps identified as impracticable to implement.
2. Based on 1a and 1b, an assessment of the overall acceptability of continued operation of Bruce A and Bruce B over the applicable period of the PSR and beyond.

5.11.1. Assessment of Defence-in-Depth

The purpose of this assessment is to address the extent to which the safety requirements of defence-in-depth are fulfilled at Bruce A and Bruce B.

IAEA publication SRS-46, Assessment of Defence in Depth for Nuclear Power Plants [29] describes a method for assessing defence-in-depth capabilities of an existing plant, including both its design features and the operational measures taken to ensure safety. A systematic identification of the required safety provisions for the siting, design, construction and operation of the plant provides the basis for assessing the comprehensiveness and quality of defence in depth at the plant. A broad spectrum of provisions, which encompass the safety features, equipment, procedures, staff availability, staff training and safety culture aspects, is considered. However, the PSR process also encompasses a systematic evaluation of the same aspects of defence-in-depth in an NPP using a different topical approach. In this context SRS-46 states the following:

> *The assessment method described in this publication is not meant to replace the other evaluations required by national or international standards. Rather, it is **intended to complement regulatory evaluations** and to provide an additional tool for a better appreciation of the defence in depth capabilities of a plant.*

Consequently, a complete DID assessment based on IAEA SRS-46 would result in duplication of the assessments conducted as part of this PSR. However, elements of DID addressed in SRS-46 can be used in developing an integrated approach to summarize DID provisions of an operating plant, findings of the Safety Factor reviews and the resulting GA and IIP from the perspective of DID.

Stated another way, the distinction between a DID assessment performed within the context of a PSR and a stand-alone DID assessment is of fundamental importance. Indeed, this distinction is explicitly recognized in both IAEA SSG-25 and CNSC REGDOC-2.3.3, which do not invoke IAEA SRS-46. In this PSR, the guidance from SRS-46 in the context of SSG-25 and REGDOC-2.3.3 has been optimized by performing an evaluation of each of the safety principles from IAEA INSAG-12 Basic Safety Principles for Nuclear Power Plants [30] using the assessments documented in the Safety Factor reports, and then integrating the findings for each level of DID.

Table 2 of SRS-46 shows the assignment of safety principles in INSAG-12 to each level of DID. This relationship has been used as the basis for establishing an approach to assessment of DID.

- Evaluate each applicable safety principle based on the current plant and the SF reviews conducted.
- For each DID level integrate results from all safety principles reviewed and the results of GA and IIP which address SF findings.

This approach provides an integrated picture of the DID features of the current plant, as well as contribution of the improvements planned in the IIP for the future.

The following process is used:

1. Establish the applicable safety principles for the DID review.
2. Define DID levels impacted for each applicable safety principle in SRS-46 (taken from INSAG-12 [30]).
3. Map each safety principle to the relevant Safety Factor reviews that have been conducted.
4. Assess the DID aspects of each safety principle in Bruce A and Bruce B design and operation at a high level. The assessment for each safety principle uses a breakdown that aligns with the levels of DID from the SRS-46 Objective Trees, as follows.
   a. Where a single level of DID is addressed within a single Objective Tree in SRS-46, there is an assessment of that specific level of DID under the safety principle. For example, in SRS-46, D-195 is addressed in three separate Objective Trees: Level 1 (Figure 30 from SRS-46), Level 2 (Figure 31 from SRS-46) and Level 3 (Figure 32 from SRS-46). Therefore, in the discussion of D-195 there are three subheadings, namely, Level 1, Level 2, and Level 3.
   b. Where multiple levels of DID are addressed within a single Objective Tree in SRS-46, there is an assessment that groups the levels of DID under the safety principle. For example, in SRS-46, O-265 is addressed in two separate Objective Trees: Levels 1 – 4 (Figure 62 from SRS-46) and Level 5 (Figure 78 from SRS-46). Therefore, in the discussion of O-265 there are two subheadings, namely, Levels 1 – 4 and Level 5.

   Compile and summarize associated evaluations from relevant Safety Factor Reports as well as the Bruce A and Bruce B Safety Reports Part 1: Plant and Site Description and Part 2: Plant Components and Systems as principal sources of information.

   Each review is common to both Bruce A and Bruce B, as the design, operation, organization and management of both plants are the same or very similar in terms of this type of assessment. Specific reference is made to unique features of a plant as appropriate.
5. Review results of SFRs for strengths, review the GA and IIP to determine those strengths and improvement initiatives that will demonstrate and further enhance alignment of Bruce A and Bruce B design and operation with the relevant safety principle.
   a. Summarize those features of the current plant design and operation that address the safety principle at a high level.
   b. Provide a list of SFR strengths, GIOs and associated CARDs included in the IIP in Part IV that further improve alignment with the safety principle.
6. Provide an overall summary integrating conclusions from each step above for each level of DID.

5.11.2. Assessment of Overall Safety

The assessment of overall safety is addressed qualitatively in terms of:

1. **Significant improvements implemented since Bruce A and Bruce B were put into operation**

   A summary of major projects undertaken to improve the physical plant to meet Power Reactor Operating Licence (PROL) conditions and to confirm safety margins as they relate to the deterministic and probabilistic safety analyses are addressed.

2. **Current IIP, major projects and initiatives driven by Asset Life Management and Ageing Management that will minimize risks associated with SSC ageing and improve safety margins**

   The current IIP, relevant current capital projects and initiatives as well as Asset Life Management Options that are planned to be implemented are discussed in terms of their contribution to safe and reliable operation. In this context, their contribution in maintaining and improving the physical plant to meet PROL conditions and to improve current safety margins as they relate to the Deterministic Safety Analysis (DSA) and Probabilistic Safety Assessment (PSA) is addressed.

3. **Compliance with regulatory dose limits as well as Bruce Power's safety goals**

   In this sub-section a summary of current DSA and PSA results that confirm conformance with the associated acceptance criteria and limits is discussed. Contribution of the major improvements covered in sub-sections 1 and 2 above will be reviewed in terms of their contribution to maintenance and improvement of DSA safety margins as well as PSA goals. Where possible, these will be addressed quantitatively rather than qualitatively.

4. **Impact of those findings that were not included for consideration in the IIP**

   In this sub-section a qualitative assessment of those findings that were assessed as impracticable will be addressed in terms of their risk reduction worth. Where possible, these will be addressed quantitatively rather than qualitatively.

5.11.3. Acceptability of Continued Operation

This section summarizes acceptability of continued operation of Bruce A and Bruce B for the 10-year PSR evaluation period based on the results of the GA and the resulting IIP, which is a living record of continuous improvement. The following is addressed:

- Completion of a comprehensive assessment of Bruce Power's current organization, governance and processes associated with all aspects of plant operation and the physical plant against the current licensing basis and modern codes and standards for Bruce A and Bruce B;
- Demonstration of the extent to which Bruce A and Bruce B design, physical plant, operation and applicable governance meet current licensing basis, associated safety

goals and fundamental safety principles within the context of defence-in-depth as well as modern codes and standards;

- Confirmation of a well developed state-of-the-art framework based on best industry practices, which continues to ensure current condition and aging of SSCs important to safety and reliability is understood and effectively managed;
- Implementation of an approach that integrates improvements planned or in-progress based on asset life management and ageing management inputs with those proposed in the IIP to mitigate SSC aging to ensure continued safe and reliable long-term operation; and
- Confirmation of the capability of Bruce Power's current organizational structure and management system, to provide the requisite tools, resources and oversight that will ensure effective execution of the IIP.

5.11.4. Document Global Assessment

The GA provides an overall review of the safety of the plant for continued operation with an extended operating life based on the integrated results from the Safety Factor reviews. The review includes the following as described in the previous sections:

- Significant PSR outcomes, including positive and negative findings (strengths and gaps);
- Analysis of interfaces, overlaps and omissions between Safety Factors and between individual negative findings;
- Classification of micro-gaps in terms of practicability and safety significance;
- The category, ranking and priority of safety improvements proposed to address negative findings;
- Justification for not pursuing certain corrective actions or safety improvements (if any) based on risk-informed analysis;
- An assessment of defence-in-depth;
- An assessment of the overall safety; and
- Justification for the overall acceptability of operation for the 10-year PSR applicability period.

5.12. Prepare Global Assessment and Integrated Implementation Plan Report

The results of the steps outlined in 5.10.3.2 and 5.11.4 above are documented in a GA and IIP Report (this Report) to include the following:

- A summary of the outcomes from the Safety Factor reports, including a list of findings indicating areas where the standards and practices considered in the PSR are not achieved, and a list of areas where they are exceeded (that is, plant strengths);
- Outcomes from the global assessment; and
- An IIP in the form of proposed list of safety improvements, including their safety significance, prioritization and timing for implementation.

Part II: Integrated Review of Safety Factor Reports

Section	Title
6	Integrated Review of Results from All Safety Factor Reports

Appendix	Title
Appendix B	Regulatory Documents, Codes and Standards Considered for Assessment

6. Integrated Review of Results from All Safety Factor Reports

The overall objective of this PSR is to conduct a review of Bruce A and Bruce B to ensure the current licensing basis will remain valid over the evaluation period and to compare against modern codes and standards and international safety expectations. This review is conducted with a view to provide input to a practicable set of improvements to be executed during the MCR in Units 3 to 8, and asset management activities to support ongoing operation of all six units, that will enhance safety and reliability to support long term operation. The look-ahead period covers a 10-year period, since there is an expectation that a PSR will be performed on approximately a 10-year cycle, given that all eight units are expected to be operated well into the future.

This section summarizes the results of the Safety Factor Report findings and identified micro-gaps, acceptable deviations and strengths, and serves as the summary of the basis for Global Assessment and development of the IIP. This summary also helped integrate all of the results summarized in Section 8 of each SFR, such that:

- All codes and standards that have been assessed by more than one Safety Factor with strengths/micro-gaps have been identified;
- For each code or standard that was assessed in multiple SFRs, those clauses that have unique and/or multiple strengths/micro-gaps have been identified;
- Micro-gaps and strengths that are same or similar across all SFRs have been identified as input for the GA; and
- Micro-gaps and strengths identified in one station but not the other have been identified for applicability as input for the GA.

This has allowed comparison of strengths/micro-gaps associated with each clause/article of a regulatory document, code or standard to ensure consistency, as well as helped in the consolidation of negative and positive findings described in Section 7. Moreover, it showed that there were no identifiable themes in the acceptable deviations that would necessitate re-classifying them as gaps.

As noted, the results summarized here are for the purpose of providing supporting information for Global Assessment and IIP development. As such, individual SFRs should be consulted for details.

The results in this section are summarized under the grouping listed in Table 1 of both the ISR and PSR Basis Documents [1] [2] recognizing the relationship amongst six main review topics as illustrated in Figure 7. This figure illustrates the PLAN-DO-CHECK-ACT principle such that:

- PLAN- Management related Safety Factors
- DO- Plant, Radiation Protection and Environment related Safety Factors
- CHECK- Safety Analysis and Performance and Feedback of OPEX related Safety Factors

- ACT- Performance and Feedback of OPEX related Safety Factors

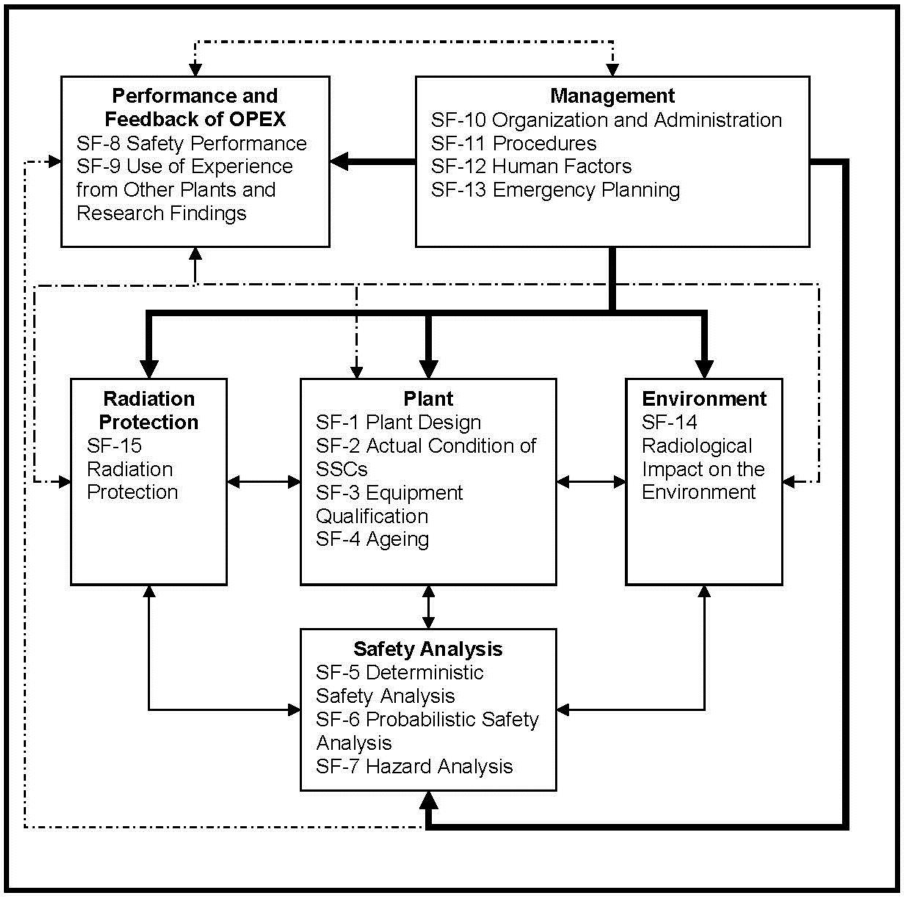

Figure 7: Integrated Review of all Safety Factor Reports

Each sub-section summarizes the results of SFRs for both Bruce A and Bruce B. To the extent possible, aspects of the summaries that are common to both stations have been combined and only those sections that require station specific information have been summarized separately.

For each SFR, results are summarized under a standard set of headings. If any modifications to the guidance provided in the PSR Basis Document have been made; they are also addressed together with the rationale for the change(s).

- Objective: This section provides the objective of the Safety Factor review as described in the PSR Basis Document- Common to both Bruce A and B unless specified otherwise.
- Scope of the Review: This section provides the review tasks performed as described in the PSR Basis Document- Common to both Bruce A and B unless specified otherwise.
- Regulatory Documents, Codes and Standards Assessed: This section summarizes the applicable regulatory documents, codes and standards that were assessed in the review

of each Safety Factor and the type of review conducted for each station. The full list of applicable regulatory documents, codes and standards that were considered for evaluation is addressed in detail in Section 3 of each Safety Factor Report, together with the final assessment types and the rationale for any changes relative to the assignment types listed in Table C-1 of [1] and [2].

- Overview of Applicable Bruce A and Bruce B Station Programs and Processes: This section summarizes the Bruce Power governance applicable to the Safety Factor review objective and tasks. Relationship amongst the relevant programs and implementing procedures are illustrated in a pictorial form common to both Bruce A and B unless specified otherwise.
- Interfaces with other Safety Factors: In this section only those interfaces of the Safety Factor in question are discussed with those under the same group of Safety Factors. There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce A ISR and Bruce B PSR. Those aspects have been addressed in each Safety Factor in detail; common to both Bruce A and B unless specified otherwise.
- Summary and Conclusions: In this section a summary of observed strengths, as well as findings based on the review tasks results is provided. Observed strengths if not common to both stations are specified accordingly. The table of key issues, which lists all the findings and presented in Section 8 of each SFR, is reproduced in this section for Bruce A and Bruce B. These strengths and findings form the basis for the global assessment and development of the IIP.

6.1. The Plant

This section summarizes the results of Safety Factors associated with the physical plant:

- SF-1 Plant Design
- SF-2 Actual Condition of SSCs
- SF-3 Equipment Qualification
- SF-4 Ageing

6.1.1. Plant Design

6.1.1.1. Objective

The objective of the review of this Safety Factor is to determine the adequacy of the design of the nuclear power plant and its documentation by assessment against modern national and international standards and practices.

6.1.1.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2]. The review of plant design (including site characteristics) includes the following tasks:

1. Review of the list of SSCs important to safety for completeness and adequacy.
2. Review to verify that design and other characteristics are appropriate to meet the requirements for plant safety and performance for all plant conditions and the applicable period of operation, including:
    a. The prevention and mitigation of events (faults and hazards) that could jeopardize safety;
    b. The application of defence-in-depth and engineered barriers for preventing the dispersion of radioactive material (integrity of fuel, cooling circuit and containment building);
    c. Safety requirements (for example, on the dependability, robustness and capability of SSCs important to safety); and
    d. Design codes and standards.
3. Identification of differences between standards met by the nuclear power plant's design (for example, the standards and criteria in force when it was built) and modern nuclear safety and design standards;
4. Review of the adequacy of the design basis documentation;
5. Review for compliance with plant design specifications;
6. Review of the safety analysis report or licensing basis documents following plant modifications and in light of their cumulative effects and updates to the site characterization;
7. Review of plant SSCs important to safety to ensure that they have appropriate design characteristics and are arranged and segregated in such a way as to meet modern requirements for plant safety and performance, including the prevention and mitigation of events that could jeopardize safety; and
8. Review of the strategy for the spent fuel storage and conduct of an engineering assessment of the condition of the storage facilities, the records management and the inspection regimes being used.

6.1.1.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 1 [17] [18] [20].

6.1.1.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Plant Design processes as identified in BP-MSM-1 Sheet 0001 under the functional area of Configuration Management Engineering. As shown in Figure 8, there are four major programs that drive plant design. These four major programs provide input to each other and are closely coupled with BP-PROG-11.01 Equipment Reliability and BP-PROG-11.04 Plant Maintenance which are covered under SF-2, SF-3 and SF-4. Interfaces with the associated procedures of BP-PROG-11.01 Equipment Reliability and BP-PROG-11.04 Plant Maintenance are not shown for simplicity, but are shown in similar figures for SF-2, SF-3 and SF-4.

The Program documents and the lower tier documents that support them are summarized in SFR 1 [17] [18] [20].

The Bruce Power programs that relate to plant design are:

- BP-PROG-10.01: Plant Design Basis Management- The objective of the plant design basis management program is to maintain the design basis and to ensure that the plant can operate safely for the full duration of the operating life of the plant. The processes contained under the elements of this program provide consistent methods for performance of the Engineering work and other activities required to meet the program objectives. This program ensures that the plant design meets safety, reliability and regulatory requirements including pressure boundary quality assurance requirements described in BP-PROG-00.04, Pressure Boundary Quality Assurance Program. Additionally, this program sets out requirements for engineering and nuclear safety analysis and documentation, such that the adequacy of the design can be demonstrated.
- BP-PROG-10.02: Engineering Change Control (ECC)- The objective of the ECC Program is to manage design changes and modifications to ensure that they are effectively defined, planned, implemented and controlled. The ECC process applies to all changes that affect design and associated documents, including:
    - New Structures, Systems, Components and Significant Tools (SSCTs);
    - Changes to existing SSCTs;
    - SSCTs to be abandoned in place, removed or demolished; and
    - Changes that affect documentation only.
- BP-PROG-10.03: Configuration Management- The objective of the Configuration Management Program is to ensure modifications to the plant, operation, maintenance and testing of the physical plant configuration is in accordance with the design requirements as expressed in the facility configuration information and to maintain this consistency throughout the operational life-cycle phase, particularly as changes are being made.
- BP-PROG-00.04: Pressure Boundary Quality Assurance Program- The PB QA Program ensures that all technical and QA requirements necessary to meet regulatory and

licence requirements related to pressure boundary are integrated into the business processes comprising Bruce Power's Management System in order to control the quality of pressure boundary activities at the company facilities for the scope of activities specified.

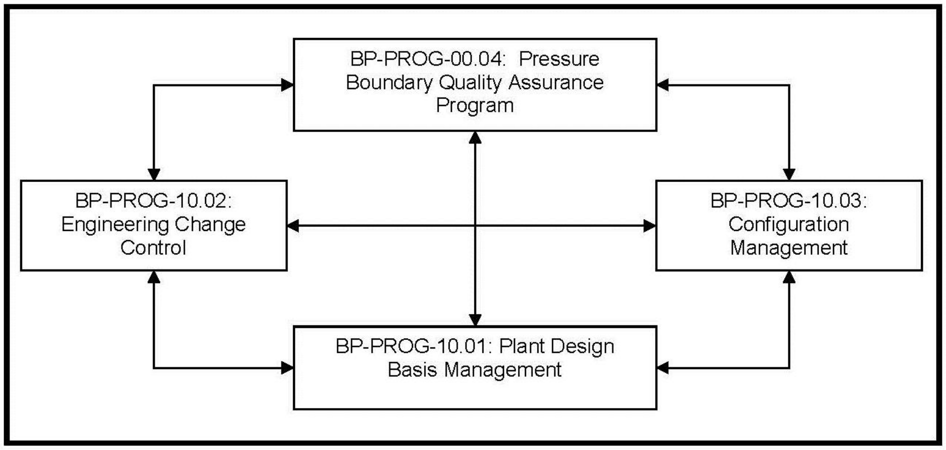

Figure 8: Overview of Applicable Bruce A Station Programs and Processes

It should be noted that all safety analysis related Safety Factors are covered under BP-PROG-10.01 Plant Design Basis Management.

6.1.1.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 1 [17] [18] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-1 with those under 'Plant' are discussed. Plant design provides the basis and technical requirements for equipment qualification and those activities associated with ensuring the actual condition of SSCs and ageing impacts remain within the design basis of the plant. This relationship is illustrated in Figure 9.

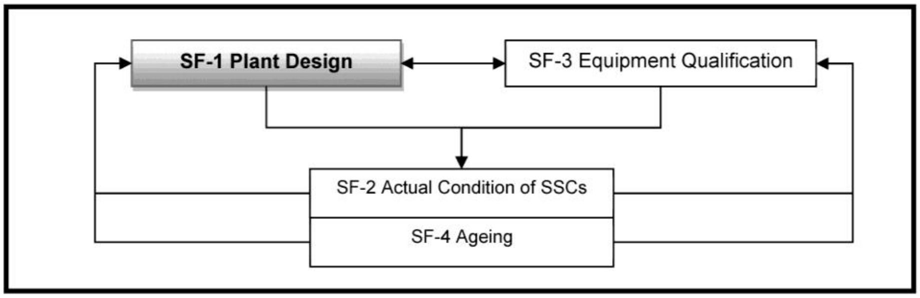

Figure 9: Safety Factor 1 Interfaces

6.1.1.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.1.1.2 are included in Section 5 of SFR 1 [17] [18] [20].

No specific strengths were identified during this review.

The key issues (or macro-gaps) arising from SFR 1 are provided verbatim in Table 1 and Table 2. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate, practicable opportunities for safety improvements are included in the IIP.

In addition, the following acceptable deviations were identified:

- CNSC REGDOC-2.5.2 (Clause 4.3.3) – Bruce A and B
- CNSC REGDOC-2.5.2 (Clause 7.6) – Bruce A and B
- CNSC REGDOC-2.5.2 (Clause 7.13) – Bruce A
- CSA N290.0-1 (Clause 4.11.2.13) – Bruce A.

These reviews concluded that overall, plant design and its management at Bruce Power meets the requirements of the Safety Factor related to plant design with the exceptions noted in Table 1 and Table 2. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The review indicates that the current and planned implementations of the programs related to plant design are adequate to support continued safe and reliable operation of Bruce A and B.

Table 1: Key Issues Identified for SFR 1 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF1-1	Safety Objectives and Concepts Event classification scheme of plant states (AOOs, DBAs, BDBAs and DECs) is not applied in the current safety analysis.	Section 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 4.2.1 REGDOC-2.5.2 - Clause 4.2.3 (Gap 1, Gap 2) REGDOC-2.5.2 - Clause 6.1 REGDOC-2.5.2 - Clause 6.4 REGDOC-2.5.2 - Clause 7.3 REGDOC-2.5.2 - Clause 7.3.2 REGDOC-2.5.2 - Clause 7.3.4 REGDOC-2.5.2 - Clause 7.4 REGDOC-2.5.2 - Clause 7.4.1 REGDOC-2.5.2 - Clause 7.5 REGDOC-2.5.2 - Clause 7.6.3 REGDOC-2.5.2 - Clause 7.13.1 REGDOC-2.5.2 - Clause 8.1 REGDOC-2.5.2 - Clause 8.1.1 REGDOC-2.5.2 - Clause 8.3.2 REGDOC-2.5.2 - Clause 8.4.1 REGDOC-2.5.2 - Clause 8.6.12 (Gap 2) REGDOC-2.5.2 - Clause 9.2 CSA N290.0 - Clause 4.2 CSA N290.0 - Clause 4.8 CSA N290.0 – Clause 4.12.4 CSA N290.0 – Clause 4.12.5
SF1-2	Safety Goals Although the results of Bruce A PRA meet the safety goal limits set up for Bruce A PRAs, they do not meet the more stringent quantitative safety goal targets set up in the requirement clause.	Section 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 4.2.2
SF1-3	Initiating Events A systematic approach to identifying a comprehensive set of postulated initiating internal and external events, including common-cause initiating events, has not been consistently applied.	Section 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 4.2.3 (Gap 2) REGDOC-2.5.2 - Clause 6.1.1 REGDOC-2.5.2 - Clause 6.6.1 REGDOC-2.5.2 - Clause 7.15.1 CSA N290.3 - Clause 10.1 CSA N290.11 – Clause 5.2.2.10

Issue Number	Macro-Gap Description	Source(s)
SF1-4	Legacy Design Analysis The original design analyses predate CSA N286.7-99.	Section 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 5.3
SF1-5	Design for Reliability Reliability requirements for some SSCs do not meet the requirements and/or safety goals.	Sections 5.3.9 and 5.7 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.6.2 CSA N290.1 - Clause 4.2.1.1 CSA N290.0 - Clause 4.5.2.1 CSA N290.0 - Clause 4.7.3 CSA N290.3 – Clause 14.1 Micro-gaps against guidance clauses: REGDOC-2.5.2 - Clause 8.4.2
SF1-6	Overpressure Protection of pressure-retaining SSCs There is not a systematic analysis of the control system capability to cope with AOOs.	Section 5.2 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.7
SF1-7	Operator Actions The current design documentation does not specifically address the timing requirements introduced in this clause.	Sections 5.3.9 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.10.4 CSA N290.11 – Clause 5.2.2.4
SF1-8	Guaranteed Shutdown State Current design documentation does not reflect required functional test frequency.	Section 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.11
SF1-9	Fire Safety Operating procedures should be developed and/or updated to incorporate the manual actions credited in the FSSA.	Section 5.3 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 9.3

Issue Number	Macro-Gap Description	Source(s)
SF1-10	Lifting and handling of large loads Identification and justification of traversing routes for large loads does not exist in current Bruce Power design documentation.	Section 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.15.3 (Gap 1,Gap 2)
SF1-11	Design Extension Conditions The current design documentation does not explicitly consider the load conditions during DECs.	Section 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.6.12 (Gap 1) REGDOC-2.5.2 - Clause 8.8 CSA N290.3 - Clauses 5.5 and 5.7
SF1-12	Electrical Power Systems Design limits are not specified for electromagnetic emissions. The DMs and OSR do not explicitly state that the SSCs employed are qualified for electromagnetic noise disturbances and mechanical vibrations. The capacity requirements and design provisions for periodic testing are not sufficiently documented. The existing safety analysis does not consider events with station blackout.	Sections 5.3.9 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.9 REGDOC-2.5.2 - Clause 8.9.2 REGDOC-2.5.2 - Clause 8.9.3 CSA N290.1 - Clause 4.7.2
SF1-13	Fuel Handling and Storage The requirement for sufficient space to accommodate the entire reactor core inventory at all times is not reflected in the design and operating documentation The radioactive sources other than the reactor core are not addressed in Part 3 of the Safety Report. A limited set of Fuel Handling System Failures is discussed in Appendix 1 and Section 3.5.5 Fuel Bay Accidents of Part 3 of the Safety Report.	Section 5.8 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.12.2 REGDOC-2.5.2 - Clause 9.1

Issue Number	Macro-Gap Description	Source(s)
SF1-14	Radiation and Environmental Protection and Mitigation The existing design documentation does not describe all necessary suitable provisions to minimize exposure, contamination, and radiological releases to the environment.	Section 5.2 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.13.3 CSA N290.2 - Clause 5.12.5 Micro-gaps against guidance clauses: REGDOC-2.5.2 - Clause 8.13 REGDOC-2.5.2 - Clause 8.13.1 (Gap 1,Gap 2) REGDOC-2.5.2 - Clause 10.1
SF1-15	Revision Changes to Stress Limit The impact on pressure boundary design governance documentation due to changes of the stress limit for “membrane longitudinal stress plus discontinuity longitudinal stress” has not been assessed.	Section 5.4 Micro-gaps against requirement clauses: ASME Section III
SF1-16	Bellows Design The impact on pressure boundary design governance due to changes to bellow design requirements has not been assessed.	Section 5.4 Micro-gaps against requirement clauses: ASME Section VIII
SF1-17	Safety Basis Report Findings Potential issues mentioned in the SBR [2] regarding changes to ASME B31.1 from 2007 to 2011 have not been addressed.	Section 5.4 Micro-gaps against requirement clauses: ASME B31.1

Table 2: Key Issues Identified for SFR 1 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF1-1	Safety Objectives and Concepts Event classification scheme of plant states (AOOs, DBAs, BDBAs and DECs) is not applied in the current safety analysis.	Sections 5.3.8, 5.3.15 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 4.2.1 REGDOC-2.5.2 - Clause 4.2.3 (Gap 1) REGDOC-2.5.2 - Clause 6.1 REGDOC-2.5.2 - Clause 6.4 REGDOC-2.5.2 - Clause 7.3 REGDOC-2.5.2 - Clause 7.3.2 REGDOC-2.5.2 - Clause 7.3.4 REGDOC-2.5.2 - Clause 7.4 REGDOC-2.5.2 - Clause 7.4.1 REGDOC-2.5.2 - Clause 7.5 REGDOC-2.5.2 - Clause 7.13.1 (Gap 2) REGDOC-2.5.2 - Clause 7.15.1 REGDOC-2.5.2 - Clause 8.1 REGDOC-2.5.2 - Clause 8.1.1 REGDOC-2.5.2 - Clause 8.3.2 REGDOC-2.5.2 - Clause 8.4.1 REGDOC-2.5.2 – Clause 9.1 (Gap 2) REGDOC-2.5.2 - Clause 9.2 CSA N290.0 - Clause 4.2 CSA N290.0 – Clause 4.12.4 CSA N290.0 – Clause 4.12.5
SF1-2	Safety Goals Although the results of Bruce B PRA meet the safety goal limits set up for Bruce B PRAs, they do not meet the more stringent quantitative safety goal targets set up in the requirement clause. The aggregate SCDF and LRF obtained by summation across all available PRA types are higher than the safety goal targets set forth in the requirement Clause 4.2.2 of CNSC REGDOC-2.5.2.	Sections 5.3.15 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 4.2.2
SF1-3	Initiating Events A systematic approach to identifying a comprehensive set of postulated initiating internal and external events, including common-cause initiating events, has not been consistently applied.	Sections 5.3.11, 5.3.12, 5.3.15 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 4.2.3 (Gap 2) REGDOC-2.5.2 - Clause 6.1.1 REGDOC-2.5.2 - Clause 6.6.1 REGDOC-2.5.2 - Clause 7.6.2 (Gap 2) CSA N290.3 - Clause 10.1 CSA N290.11 – Clause 5.2.2.10

Issue Number	Macro-Gap Description	Source(s)
SF1-4	Legacy Design Analysis Many of the original design analyses were produced using tools that predated N286.7-99.	Section 5.3.15 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 5.3
SF1-5	Design for Reliability Reliability requirements for some SSCs do not meet the requirements and/or safety goals.	Sections 5.3.8, 5.3.15 and 5.7 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.6.2 (Gap 1) CSA N290.0 - Clause 4.7 CSA N290.0 - Clause 4.11.2.13
SF1-6	Systematic Analysis of Overpressure Protection of pressure-retaining SSCs There is not a systematic analysis of the control system capability to cope with AOOs.	Sections 5.2 and 5.3.15 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.7
SF1-7	Operator Emergency Response Requirement related to sufficiency of staff credited with performing contingency activities on outage heat sinks has not been demonstrated to be met.	Sections 5.3.12 and 5.6 Micro-gaps against requirement clauses: CSA N290.11 – Clause 5.2.2.4
SF1-8	Guaranteed Shutdown State (GSS) Current design documentation does not reflect required functional test frequency for the equipment associated with GSS.	Sections 5.3.15 and 5.4 Micro-gaps against guidance clauses: REGDOC-2.5.2 - Clause 7.11
SF1-9	Timing of Operator Actions The current safety analysis does not meet the timing requirements of operator actions of 30 min and 1 h. In addition, the current design documentation does not reflect the requirement for long-term services for emergency support systems.	Sections 5.3.9, 5.3.15, 5.4, and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.10 REGDOC-2.5.2 - Clause 8.10.4 CSA N290.1 – Clause 4.3.1.4
SF1-10	Lifting and handling of large loads Identification and justification of traversing routes for large loads, and analysis to justify safe operations when considering the drop of large loads does not exist in current Bruce Power design documentation.	Sections 5.3.15 and 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.15.3 (Gap 1,Gap 2)

Issue Number	Macro-Gap Description	Source(s)
SF1-11	Design Extension Conditions The current design documentation does not explicitly consider the load conditions on containment during DECs.	Sections 5.3.15 and 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.6.12
SF1-12	Electrical Power Systems Design limits are not specified for electromagnetic emissions. The design manuals and OSR do not explicitly state that the SSCs employed are qualified for electromagnetic noise disturbances and mechanical vibrations. The capacity requirements and design provisions for periodic testing are not sufficiently documented. The existing safety analysis does not consider events with station blackout.	Sections 5.3.9 5.3.15 and 5.6 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.9 REGDOC-2.5.2 - Clause 8.9.2 REGDOC-2.5.2 - Clause 8.9.3 CSA N290.1 - Clause 4.7.2
SF1-13	Fuel Handling and Storage The requirement for sufficient space to accommodate the entire reactor core inventory at all times is not reflected in the design and operating documentation. The radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, are not addressed in Part 3 of the Safety Report.	Sections 5.3.15, 5.4 and 5.8 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.12.2 REGDOC-2.5.2 - Clause 9.1 (Gap 1)
SF1-14	Radiation and Environmental Protection and Mitigation The existing design documentation does not describe all necessary suitable provisions to minimize exposure, contamination, and radiological releases to the environment.	Sections 5.2, 5.3.10, 5.3.15 and 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.13.1 (Gap 1) REGDOC-2.5.2 - Clause 8.13.3 CSA N290.2 - Clause 5.12.5 Micro-gaps against guidance clauses: REGDOC-2.5.2 - Clause 8.13 REGDOC-2.5.2 - Clause 8.13.1 (Gap 2) REGDOC-2.5.2 - Clause 10.1
SF1-15	Seismic Instrumentation Earthquake monitoring instrumentation is not installed in the plant.	Sections 5.3.3 and 5.3.15 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.13 CSA N289.1 – Clauses 6.5.6.3 and 6.5.6.4

Issue Number	Macro-Gap Description	Source(s)
SF1-16	Seismic Qualification Documentation Governing and implementing documents for seismic qualification do not consistently indicate the application of CSA N289 series. The more recent site investigations documented in the Probabilistic Seismic Hazard Assessment are not reflected in the design documentation.	Sections 5.3.3, 5.3.15, and 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.13.1 (Gap 1) Micro-gaps against guidance clauses: CSA N289.1 – Clause 3.1
SF1-17	Revision Changes ASME Section III There is no evidence that pressure boundary design governance documentation and safety margins has been reviewed for impact of changes in Stress Limits, Bolting S_m Values, Stress Indices for Straight Pipe, Branch Connections and Load Limit values.	Section 5.3.17 and 5.4 Micro-gaps against requirement clauses: ASME Section III
SF1-18	Revision Changes to Pressure Boundary Design Requirements Pressure boundary design governance documentation and safety margins have not been reviewed for impact of new requirements introduced with the latest revisions of CSA N285.0 and changes in ASME Section VIII.	Sections 5.3, 5.3.17 and 5.4 Micro-gaps against requirement clauses: ASME Section VIII
SF1-19	Barriers for Containment Penetrations The safety significance of identified differences between the current design documentation and the requirements of CSA N290.3-11, Annex A has not been assessed.	Sections 5.3.11 and 5.4 Micro-gaps against requirement clauses: CSA N290.3 Clause A.2.3, CSA N290.3 Clause A.2.5, CSA N290.3 Clause A.3.1 CSA N290.3 Clause A.3.4
SF1-20	Special Safety System Requirements There are documented exceptions for design of special safety system components such that the most likely failure modes are not in the failsafe direction. There remains some instances where the failure mode is unsafe and the operator must monitor or test SDS availability. Bruce B design includes sharing of special safety systems without justification that such sharing contributed to enhanced safety as required by CNSC REGDOC-2.5.2 clause 7.6.5.2.	Sections 5.3.8, 5.3.9, 5.3.15 and 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 7.6.3 REGDOC-2.5.2 – Clause 7.6.5.2 CSA N290.0 - Clause 4.8 CSA N290.1 – Clause 4.2.6

Issue Number	Macro-Gap Description	Source(s)
SF1-21	Pressure Boundary Quality Assurance Program Deficiencies Implementation of certain elements of BP-PROG-00.04 were found ineffective. Some program elements do not meet implementing process pressure boundary quality assurance requirements.	Sections 4.1, 4.1.1, 5.4, 7.2.3, 7.2.4, and 7.2.5
SF1-22	Emergency Support Facilities The Bruce B design does not provide an onsite emergency facility (or facilities) that are separate from the plant control rooms which include a Safety Parameter Display System (SPDS) similar to those in the MCR and the SCA.	Sections 5.2 and 5.3.15 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.10.3
SF1-23	Emergency Heat Removal System Since Bruce B emergency heat removal function is provided by more than one system; it cannot be confirmed that the same function will be available during DECs, if required.	Sections 5.3.15 and 5.4 Micro-gaps against requirement clauses: REGDOC-2.5.2 - Clause 8.8 (Gap 1, Gap 2)
SF1-24	Tracking Licence Concessions Bruce Power should establish a controlled, centralized and accessible company database available to support design activities	Sections 5.3.16 and 5.4 Micro-gaps against requirement clauses: ANSI/NIRMA CM 1.0-2007 – Section 3.2

6.1.2. Actual Condition of Systems, Structures, and Components

6.1.2.1. Objective

The objective of the review in this Safety Factor is to determine the actual condition of Systems, Structures and Components (SSCs) important to safety and whether it is adequate for them to meet their design requirements. In addition, the review should confirm that the condition of SSCs is properly documented.

6.1.2.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks will include examination of the following aspects for the selected SSCs:

1. Existing or anticipated ageing processes;

2. Operational limits and conditions;
3. Current state of the SSC with regard to its obsolescence;
4. Implications of changes to design requirements and standards on the actual condition of the SSC since the plant was designed or since the last PSR (for example, changes to standards on material properties);
5. Plant programs that support ongoing confidence in the condition of the SSC;
6. Significant findings from tests of the functional capability of the SSC;
7. Results of inspections and/or walkdowns of the SSC;
8. Maintenance and validity of records;
9. Evaluation of the operating history of the SSC;
10. Dependence on obsolescent equipment for which no direct substitute is available;
11. Dependence on essential services and/or supplies external to the plant;
12. The condition and operation of spent fuel storage facilities and their effect on the spent fuel storage strategy for the nuclear power plant; and
13. Verification of the actual state of the SSC against the design basis.

6.1.2.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 2 [17] [20].

6.1.2.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Equipment Qualification processes. The fundamental objective and driver for actual condition of the SSCs important to safety is to maintain the validity of the current design basis and to ensure that it will remain valid in the future as prescribed in the PROL. This requires maintaining the functional reliability and structural integrity of SSCs important to safety as described in the design and the supporting deterministic and probabilistic safety analyses which are integrated in Bruce A and Bruce B operation through BP-PROC-00363 Nuclear Safety Assessment and BP-PROC-00786 Margin Management. In this context, SF-2 is strongly coupled with SF-1, SF-3, SF-4, as well as SF-5, SF-6 and SF-7. This is illustrated in Figure 10 where BP-PROG-11.01 and BP-PROG-11.04, the implementing programs for SF-2, are linked to BP-PROG-10.01, BP-PROG-10.02, BP-PROG-10.03, the implementing programs for SF-1, SF-3, SF-4, SF-5, SF-6 and SF-7 covering the plant design and safety analysis.

There are other programs which have not been included for simplicity but support those in Figure 10:

- BP-PROG-11.02: On-Line Work Management Program
- BP-PROG-11.03: Outage Work Management
- BP-PROG-12.02: Chemistry Management
- BP-PROG-12.03: Nuclear Fuel Management

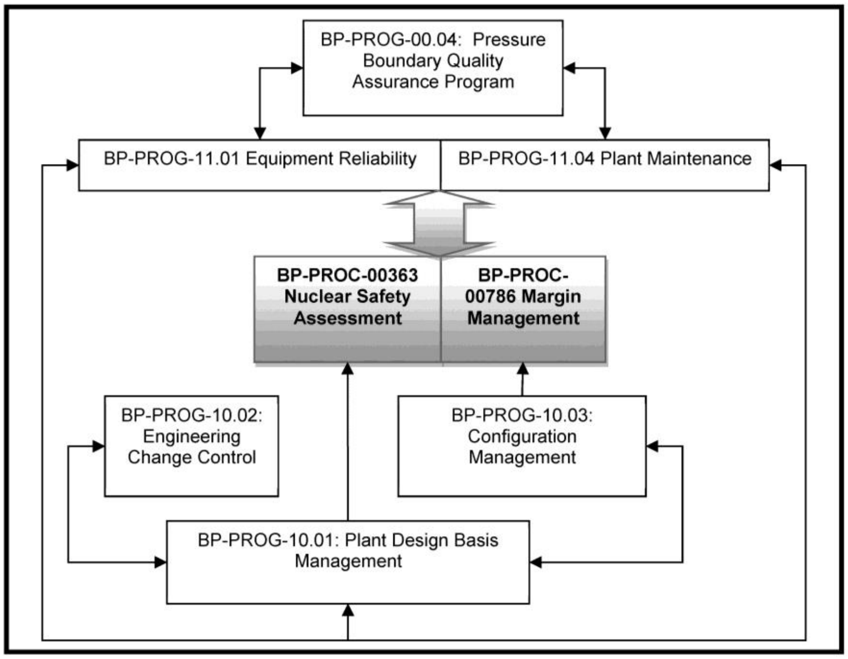

Figure 10: Safety Factor 2 Supporting Programs

BP-PROC-00786 Margin Management describes how Bruce Power manages Design and Operating Margins, fulfilling the following main objectives:

- Supporting safe and reliable plant operation.
- Ensuring plant equipment configuration and performance are consistent with design and licensing requirements.
- Conducting day-to-day operations reflecting consideration of design and operating margins.

This Margin Management document is aligned with the structure described in INPO document 09-003, Excellence in the Management of Design and Operating Margins.

BP-PROC-00363 Nuclear Safety Assessment defines the elements, functional requirements, implementing procedures and key responsibilities associated with the Nuclear Safety Assessment (NSA) process. The objective of NSA is to ensure that all necessary nuclear safety requirements are defined for the actual or proposed design of the plant throughout the design modification process or in addressing emergent issues (e.g., plant aging) that may affect the Design Basis or the Safety Report Basis.

The implementing documents relevant to Safety Factor 2 are listed in Section 4 of SFR 2 [17] [20].

As shown in Figure 11, one unique aspect of SF-2 and the overall PSR process is its integration with Bruce Power's long-term strategy for safe and reliable operation of Bruce A and Bruce B. Bruce Power has established the Asset Life Projection and Options (ALPO) process described in BP-PROC-00899 Asset Life Projections and Options and BP-PROC-00936 Asset Management Planning. Furthermore, a description of the overall Asset Management process, including a comparison of the methods using Condition Assessment Reports (CARs) and Asset Management Options Templates (AMOTs) and how current Bruce Power procedures and processes support each clause of REGDOC-2.6.3, is provided in References [26], [27], and [28].

The objective of BP-PROC-00899 is to provide an input to the Strategic Planning process and provide the required options to manage the asset to 2043 and beyond and to define the process of developing and revising an ALPO document.

An ALPO will achieve the following:

- Establish the projected end of life based on the current condition of the SSCs.
- Identify the Mitigation Options to reach component end of life based on the ARDMs (Age Related Degradation Mechanisms) and/or obsolescence issues.
- Identify the activities to maintain the asset and the health of the maintenance and surveillance program(s).
- Identify and provide recommended numbers and rationale to include the component or sub-components as part of the Strategic Spares set.

The objective of BP-PROC-00936 is to select and approve Asset Management options to achieve a resource leveled, integrated Asset Management Plan that will provide safe, reliable long term operation in alignment with corporate strategic and business planning objectives.

In this context Bruce Power's strategy is to complete any required work in normal outages but where this is not possible, in special outages such that MCR will focus on replacement of the critical life limiting components, i.e., Fuel Channels, Feeders and Steam Generators and associated enabling work. Asset Management scope will be considered within the MCR outage window if the associated work requires significant field time (>90 days), or a defueled/dewatered state or nominal case End of Life (EOL) falls in the Refurbishment outage window.

This approach described in Section 5.4 ensures that the actual condition of SSCs is maintained within their current design and operating envelope as stipulated in the PROL and LCH at all times. As a result, MCR scope which meet the criteria described in Section 5.4 will be included in the IIP based on its timing.

It should also be noted that in Figure 11 Safety Analysis and Safety Basis Review has not been included for simplicity. See Figure 10 for an integrated view of all elements.

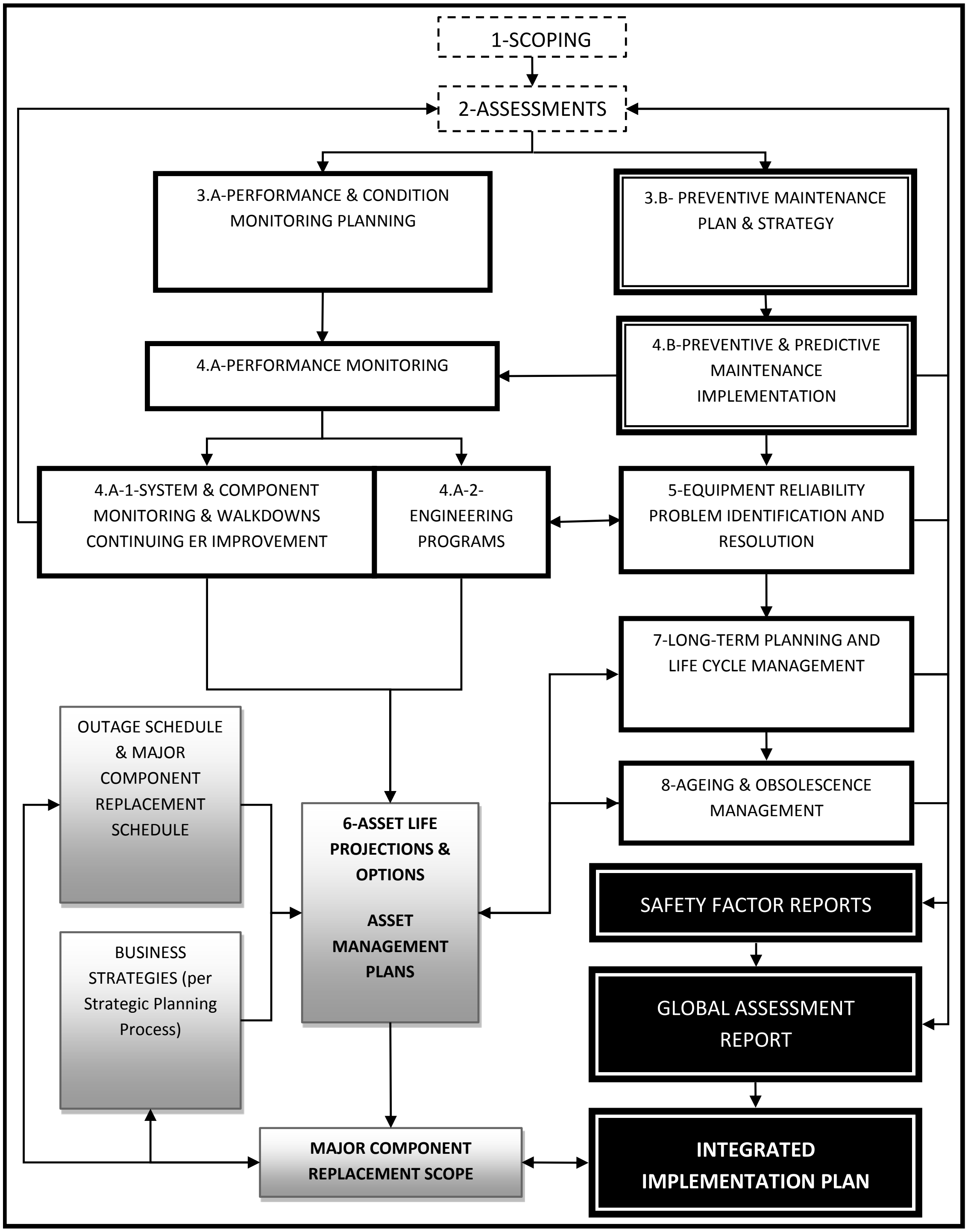

Figure 11: Relationship Between Equipment Reliability, Maintenance, Asset Life Management, MCR and IIP

Figure 11 Legend:

APPLICABLE BRUCE POWER GOVERNANCE in SUPPORT OF Figure 11

1- SCOPING

BP-PROC-00778 Scoping and Identification of Critical SSCs
BP-PROC-00584 PASSPORT Equipment Data Management
BP-PROC-00666 Component Categorization (includes SPVs- Single Point Vulnerabilities)
BP-PROC-00169 Safety-Related System List
DPT-RS-00012 Systems Important to Safety Decision Methodology

2-ASSESSMENTS

BP-PROC-00014 Technical Operability Evaluation
BP-PROC-00363 Nuclear Safety Assessment
BP-PROC-00383 Performance and Condition Assessment
BP-PROC-00498 Condition Assessment of Generating Units in Support of Life Extension
BP-PROC-00532 Critical and Strategic Spares
BP-PROC-00534 Technical Basis Assessment
BP-PROC-00789 Maintenance Strategy
BP-PROC-00849 Aggregate Risk Assessment and Monitoring
DIV-ENG-00004 Engineering Evaluations

3.A -PERFORMANCE & CONDITION MONITORING PLANNING

DPT-PE-00008 System and Component Performance Monitoring Plans

4.A-1-SYSTEM & COMPONENT MONITORING & WALKDOWNS

BP-PROC-00779 Continuing Equipment Reliability Improvement
BP-PROC-00781 Performance Monitoring
BP-PROC-00863 Engineering Programs Health Reporting
DPT-PE-00009 System and Component Performance Monitoring Walkdowns
DPT-PE-00010 System Health Reporting
DPT-PE-00011 Component Health Reporting

4.A-2-ENGINEERING PROGRAMS
NK21-PIP-21100-00001 CSA N287.7-08 Periodic Inspection Program for Bruce NGS A Concrete Containment Structures and Appurtenances (Excluding Vacuum Building)
NK21-PIP-25100-00001 CSA N287.7-08 Periodic Inspection Program for Bruce NGS A Vacuum Building
NK21-PIP-03641-00001 Bruce NGS A N285.4 Periodic Inspection Plan, Nuclear Components-General
NK21-PIP-03641-00004 Bruce NGS A N285.4 Periodic Inspection Plan for Nuclear Components-Unit 0
NK21-PIP-03641-00003 Bruce NGS A N285.4 Periodic Inspection Plan, Nuclear Components-Unit 4
NK21-PIP-03641-00002 Bruce NGS A N285.4 Periodic Inspection Plan for Nuclear Components-Unit 3
NK21-PIP-03642-00001 Bruce NGS A N285.5 Periodic Inspection Plan, for Unit 0A and Units 1 to 4 Containment Components
NK29-PIP-21100-00001 CSA N287.7-08 Periodic Inspection Program for Bruce NGS B Concrete Containment Structures and Appurtenances (Excluding Vacuum Building)
NK29-PIP-25100-00001 CSA N287.7-08 Periodic Inspection Program for Bruce NGS B Vacuum Building
NK29-PIP-03641.2-00001 Bruce NGS B N285.4 Periodic Inspection Plan for Unit 5
NK29-PIP-03641.2-00002 Bruce NGS B N285.4 Periodic Inspection Plan for Unit 6
NK29-PIP-03641.2-00003 Bruce NGS B N285.4 Periodic Inspection Plan for Unit 7
NK29-PIP-03641.2-00004 Bruce NGS B N285.4 Periodic Inspection Plan for Unit 8
NK29-PIP-03642-00001 Bruce NGS B N285.5 Periodic Inspection Plan, for Unit 0 and Units 5 to 8 Containment Components
BP-PROC-00217 Measuring and Test Equipment Calibration Program Requirements
BP-PROC-00267 Management of Steam Generator and Preheater Tube Integrity
BP-PROC-00268 Safety System Testing (SST) Program Procedure
BP-PROC-00334 Periodic Inspection
BP-PROC-00361 In-Service Testing and Inspection to Satisfy CAN/CSA N287.7-08 Requirements
BP-PROC-00387 Plant Inspection
BP-PROC-00825 Buried Piping Inspection Program
BP-PROC-00893 Fuel and Fuel Channel Program
BP-PROC-00923 Pipe Wall Thinning - FAC
SEC-RE-00017 Motor Program
SEC-ME-00008 Heat Exchangers
SEC-ME-00010 Inspection and Monitoring Once-Through Service Water Systems

3.B- PREVENTIVE MAINTENANCE PLAN & STRATEGY
BP-PROC-00457 Development and Approval of Predefined
BP-PROC-00534 Technical Basis Assessment
BP-PROC-00694 Maintenance Procedure Development and Revision
BP-PROC-00695 Maintenance Program and Activities
BP-PROC-00696 Maintenance Organization
BP-PROC-00699 Maintenance Work
BP-PROC-00789 Maintenance Strategy

4-PREVENTIVE & PREDICTIVE MAINTENANCE
BP-PROC-00780 Preventive Maintenance Implementation
BP-PROC-00284 Predictive Maintenance
BP-PROC-00603 Preventative Maintenance Program Just In Time Review Process

5-EQUIPMENT RELIABILITY PROBLEM IDENTIFICATION AND RESOLUTION
BP-PROC-00782 Equipment Reliability Problem Identification and Resolution
BP-PROC-00496: Trouble Shooting Plant Equipment

6-ASSET LIFE PROJECTIONS & OPTIONS
BP-PROC-00899 Asset Life Projections and Options
BP-PROC-00936 Asset Management Planning

7-LONG-TERM PLANNING AND LIFE CYCLE MANAGEMENT
BP-PROC-00783 Long-Term Planning and Life Cycle Management
BP-PROC-00400 Life Cycle Management for Critical SSCs

8-AGEING & OBSOLESCENCE MANAGEMENT
BP-PROC-00532 Critical and Strategic Spares
BP-PROC-00533 Obsolescence Management

6.1.2.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 SFR 2 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section, the interfaces of SF-2 with those under 'Plant' are discussed. Programs, processes and documentation under SF-1 and SF-3 provide the requirements and the bases for activities in ensuring actual conditions of SSCs important to safety are within the design basis. Outputs of the activities associated with SF-2 and SF-4 inform and provide feedback to each other. Results of the activities associated with SF-2 are fed back to those programs and processes under SF-1 and SF-3 for any follow-up and improvement opportunities. This relationship is illustrated in Figure 12.

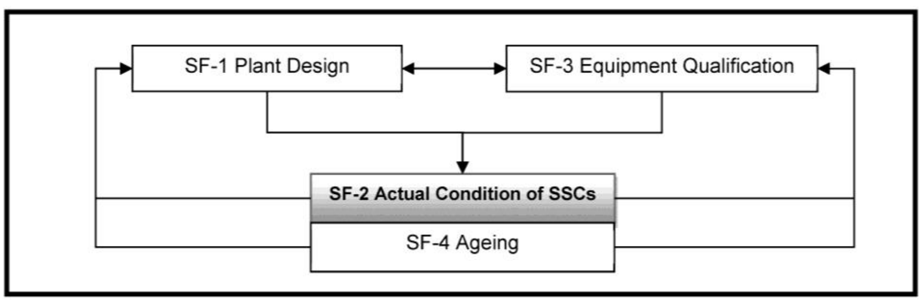

Figure 12: Safety Factor 2 Interfaces

6.1.2.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.1.2.2 are included in Section 5 of SFR 2 [17] [20]. These reviews concluded that overall, Bruce Power meets the requirements of the Safety Factor related to actual condition of the SSCs.

Strengths identified during this review for Bruce A are as follows. These strengths are considered as applicable to Bruce B:

- The conditions of the U014 and U058 SSCs are now tracked in SHRs. Bruce Power continues to improve and streamline the SHR processes as part of ageing and asset management, integrating these improvements with their anticipated obsolescence, testing, inspection and maintenance programs.
- Bruce Power's preventive maintenance implementation is a station priority. The station management team monitors implementation and leaders enforce accountability.

There were no key issues arising from SFR 2 for either Bruce A or Bruce B. The following observations were made with respect to improvement opportunities previously identified:

- There were four potential improvement opportunities described in the interim PSR [31], although none required a direct IIP item.
- The condition of the SSCs of Units 018 was assessed in [32]. A number of issues have been identified, but most are of low significance and are being tracked following the well-established Bruce Power managed processes, such as System and Component Health Reporting. The SHRs, which initiate and track projects that improve the SSC conditions, are being implemented in line with their priority as determined by the SPHC.
- Fitness for service and estimated remaining life has been assessed and is documented in the LCMPs within the Asset Management program. A number of SSCs will require replacement within the timeframe covered by this PSR. Replacement is being tracked following the well-established Bruce Power managed processes, such as System and Component Health Reporting.

Bruce Power recognizes that a significant improvement in the station equipment health is a major contributor to achieving strong safety and successful business plan performances going forward as there will be fewer unplanned, forced outages and increasingly more predictable operations. Equipment health initiatives beyond those discussed herein are planned so the stations are positioned to achieve long-term equipment reliability and plant health.

Overall, Bruce Power meets the requirements of the Safety Factor related to actual condition of the SSCs. The review demonstrates that the current implementation of the programs related to condition assessments ensure that Bruce Power is aware of the condition of the SSCs at Bruce A and Bruce B and has implemented measures to ensure that SSCs remain fit for service and meet regulatory requirements during the 10-year period covered by this PSR. Implementation and continuous improvement of the current programs and procedures, as well as the planned physical improvements in place, will ensure long-term safe and reliable operation of SSCs important to safety beyond the current assessment period.

6.1.3. Equipment Qualification

6.1.3.1. Objective

The objective of the review in this Safety Factor is to determine whether equipment important to safety is qualified (including for environmental conditions) and whether this qualification is being maintained through an adequate program of maintenance, inspection and testing that provides confidence in the delivery of safety functions.

6.1.3.2. Scope of Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. The review of equipment qualification will include an assessment of the effectiveness of the plant's equipment qualification program. This program should ensure that plant equipment (including cables) is capable of fulfilling its safety functions for the period until at least the next PSR. The review will also cover the requirements for performing safety functions while subject to the environmental conditions that could exist during both normal and predicted accident conditions. These include seismic conditions, vibration, temperature, pressure, jet impingement, electromagnetic interference, irradiation, corrosive atmosphere and humidity, fire (for example, a hydrogen fire) and combinations thereof and other anticipated events. The review will also consider the effects of ageing degradation of equipment during service and of possible changes in environmental conditions during normal operation and predicted accident conditions since the program was devised;
2. Although many parties (such as designers, equipment manufacturers and consultants) will be involved in the equipment qualification process, the operating organization has the ultimate responsibility for the development and implementation of an adequate plant specific equipment qualification program. The following aspects of implementation of the program will be covered:
    a. Assess if qualification of plant equipment important to safety has been formalized using a process that includes generating, documenting and retaining evidence that equipment can perform its safety functions during its installed service life;
    b. Confirm if this is an ongoing process, from its design through to the end of its service life; and
    c. Assess if the process takes into account plant and equipment ageing and modifications, equipment repairs and refurbishment, equipment failures and replacements, any abnormal operating conditions and changes to the safety analysis.
3. The review of equipment qualification will consider:
    a. Whether installed equipment meets the qualification requirements;
    b. The adequacy of the records of equipment qualification;
    c. Procedures for updating and maintaining qualification throughout the service life of the equipment;
    d. Procedures for ensuring that modifications and additions to SSCs important to safety do not compromise their qualification;

e. Surveillance programs and feedback procedures used to ensure that ageing degradation of qualified equipment remains insignificant;

f. Monitoring of actual environmental conditions and identification of ‘hot spots’ of high activity or temperature; and

g. Protection of qualified equipment from adverse environmental conditions.

6.1.3.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 3 [17] [20].

6.1.3.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Equipment Qualification processes. As described in Section 4 and Appendix A of SFR 3 [17] [20] the Equipment Qualification process is well defined in a number of procedures and supporting documentation, such as design guides and reports on various topics. Bruce A and Bruce B Environmental Qualification and Seismic Qualification are governed under BP-PROG-10.01 Plant Design Basis Management and are implemented by BP-PROC-00335 Design Management. BP-PROG-10.01 interfaces with other programs which have a role in Equipment Qualification, including BP- BP-PROG-11.01 Equipment Reliability and BP-PROG-11.04 Plant Maintenance. These programs collectively address the design, procurement of replacement or new qualified equipment and the monitoring of qualified equipment to preserve the qualification for the life of the station as illustrated in Figure 13.

In each procedure, there are interface links to other supporting station programs and procedures that have an important bearing on preserving the equipment qualification, such as procurement, engineering change control and condition monitoring. These are supported by a robust self-assessment and audit process to examine the various activities involved in maintaining the equipment qualification for the life of the plant in addition to the activities carried through BP-PROG-11.01 Equipment Reliability and BP-PROG-11.04 Plant Maintenance.

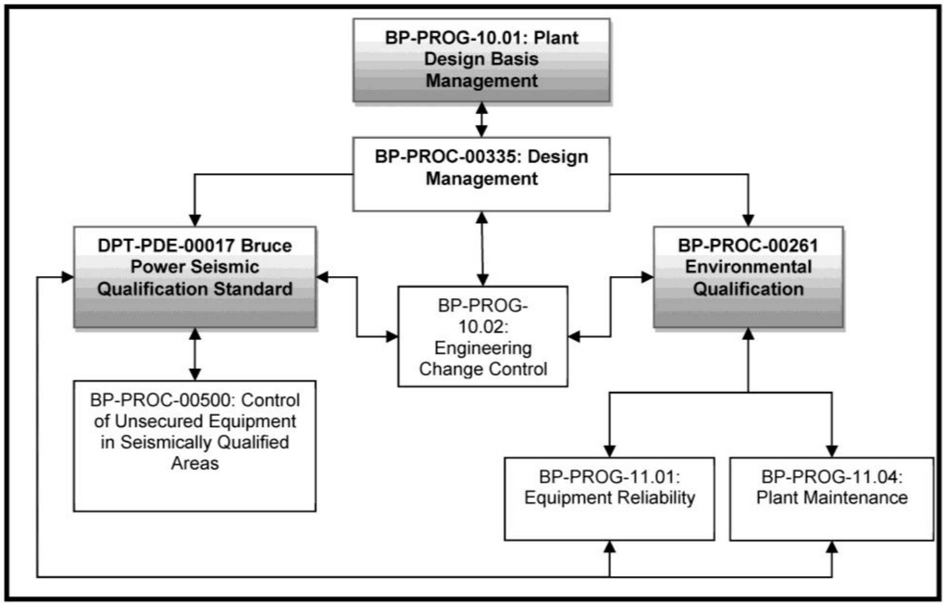

Figure 13: Overview of Governance for Equipment Qualification

6.1.3.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 3 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-3 with those under ‘Plant’ are discussed. Technical basis for equipment qualification is driven via SF-1 and technical requirements for equipment qualification and those activities associated with ensuring the actual condition of SSCs and ageing impacts remain within the design basis of the plant is provided by the programs, processes and documentation under SF-3. Results of the activities associated with SF-2 and SF-4 are fed back to those programs and processes under SF-3 for any follow-up and improvement opportunities. This relationship is illustrated in Figure 14.

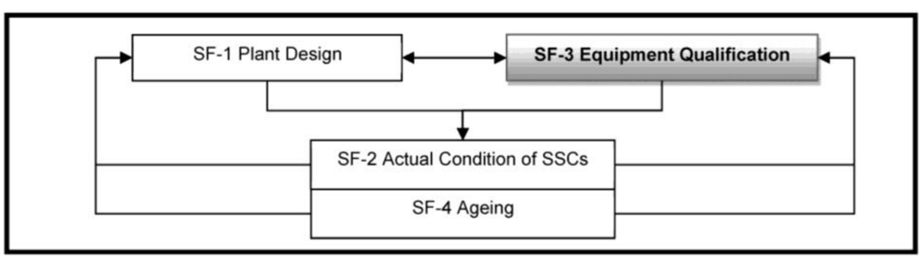

Figure 14: Safety Factor 3 Interfaces

6.1.3.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.1.3.2 are included in Section 5 of SFR 3 [17] [20]. Based on the reviews conducted, it was concluded that both Bruce A and Bruce B comply with the requirements of the most recent codes and standards for environmental qualification and seismic qualification except for the micro-gaps identified below in Table 3 due to recent changes to the N289 series of standards. Practicable improvements to close the identified micro-gaps will support compliance similar to those required for modern plants. It was also concluded that the current equipment qualification process can be sustained for the life of the plant.

Two strengths were identified during this review considered as applicable to both Bruce A and Bruce B, as follows:

- The quality of the programmatic documents (i.e., programs and procedures) for the equipment qualification process is very good, with interfaces with other station procedures well identified, recent revisions and updating for most procedures, and incorporation of issues identified in audits and self-assessments.
- The IAEA OSART review of Bruce B completed in 2015 reviewed all aspects of the environmental qualification program and recognized its overall implementation as “good performance”. Therefore, the management of the EQ program is a strength.

The key issues (or macro-gaps) arising from SFR 3 for Bruce B are provided verbatim in Table 3. There were no key issues identified for Bruce A. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate, practicable opportunities for improvement are included in the IIP. Applicability of these issues to Bruce A is also considered as part of Global Assessment.

In addition, the following acceptable deviation was identified:

- CSA N289.1-08 (Section 6) – Bruce B.

Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The review indicates that the current and planned implementations of the programs related to equipment qualification are adequate to support continued safe and reliable operation of Bruce A and B.

Table 3: Key Issues Identified for SFR 3 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF3-1	A periodic evaluation to demonstrate readiness to cope with the potential consequences of a beyond design basis seismic event once every 10 years, as a minimum, has not been done.	Section 5.1 Micro-gaps against requirement clauses: CSA N289.1-08 – Clause 5.3.11

Issue Number	Macro-Gap Description	Source(s)
SF3-2	Earthquake monitoring instrumentation that would provide accurate earthquake records to confirm that the plant is fit for continued operation following an earthquake is not installed in the plant.	Section 5.1 Micro-gaps against requirement clauses: CSA N289.1-08 – Clauses 6.5.6.3 and 6.5.6.4
SF3-3	An investigation of the potential for a seismic seiche and consequent surges along the shore that could affect the safety of the nuclear power plant has not been done.	Section 5.1 Micro-gaps against requirement clauses: CSA N289.2-10 – Clause 4.4.2.2
SF3-4	A free field accelerometer has not been installed on the site to confirm that a seismic event has occurred.	Section 5.1 Micro-gaps against requirement clauses: CSA N289.5-12 – Clause 4.1.1.3
SF3-5	The governing and implementing documents for seismic qualification do not consistently indicate the application and licensing status of the CSA N289 series of standards. The reporting and recording requirements for earthquake events and the more recent site investigations documented in the Probabilistic Seismic Hazard Assessment are not reflected in the seismic implementing procedures.	Section 5.1 CSA N289.1-08 – Clause 5

6.1.4. Ageing

6.1.4.1. Objective

The objective of the review of this Safety Factor is to determine whether ageing aspects affecting SSCs important to safety are being effectively managed and whether an effective ageing management program is in place so that all required safety functions will be delivered for the design lifetime of the plant and, if it is proposed, for long term operation.

6.1.4.2. Scope of Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. The following programmatic and technical aspects of the ageing management program are addressed:

a. The timely detection and mitigation of ageing mechanisms and/or ageing effects;

b. The comprehensiveness of the program, i.e., does it address all SSCs important to safety?

c. The effectiveness of operating and maintenance policies and/or procedures for managing the ageing of replaceable components;

d. Evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs important to safety;

e. Management of the effects of ageing on those parts of the nuclear power plant that will be required for safety when the nuclear reactor has ceased operation, for example the spent fuel storage facilities;

f. Performance indicators;

g. Record keeping.

2. The review addresses the following technical aspects:

a. Ageing management methodology;

b. The operating organization’s understanding of dominant ageing mechanisms and phenomena, including knowledge of actual safety margins;

c. Availability of data for assessing ageing degradation, including baseline data and operating and maintenance histories;

d. Acceptance criteria and required safety margins for SSCs important to safety;

e. Operating guidelines aimed at controlling and/or moderating the rate of ageing degradation;

f. Methods for monitoring ageing and for mitigation of ageing effects;

g. Awareness of the physical condition of SSCs important to safety and any features that could limit service life;

h. Understanding and control of ageing of all materials (including consumables, such as lubricants) and SSCs that could impair their safety functions; and

i. Obsolescence of technology used in the nuclear power plant.

6.1.4.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 4 [17] [20].

6.1.4.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Ageing management processes. Ageing management for Bruce A and Bruce B is governed by a cross-functional collection of governance documents that is mostly centered in the Equipment Reliability (ER) program which is defined in BP-PROG-11.01. Bruce Power's Aging Management Roadmap, which follows the PLAN-DO-CHECK-ACT approach in REGDOC-2.6.3, is provided in BP-PROC-00783, Long Term Planning and Life Cycle Management, is reproduced in Figure 15 and includes the Bruce Power programs and procedures relevant to plant ageing. The Bruce Power programs and procedures relevant to plant ageing are identified in Section 4 of SFR 4 [17] [20] and have already been identified in Section 6.1.2.4.

Figure 15 demonstrates a well integrated ageing management process for long-term safe and reliable operation of Bruce A and Bruce B. Higher level integration of ageing with plant life management programs and activities is illustrated previously in Figure 11.

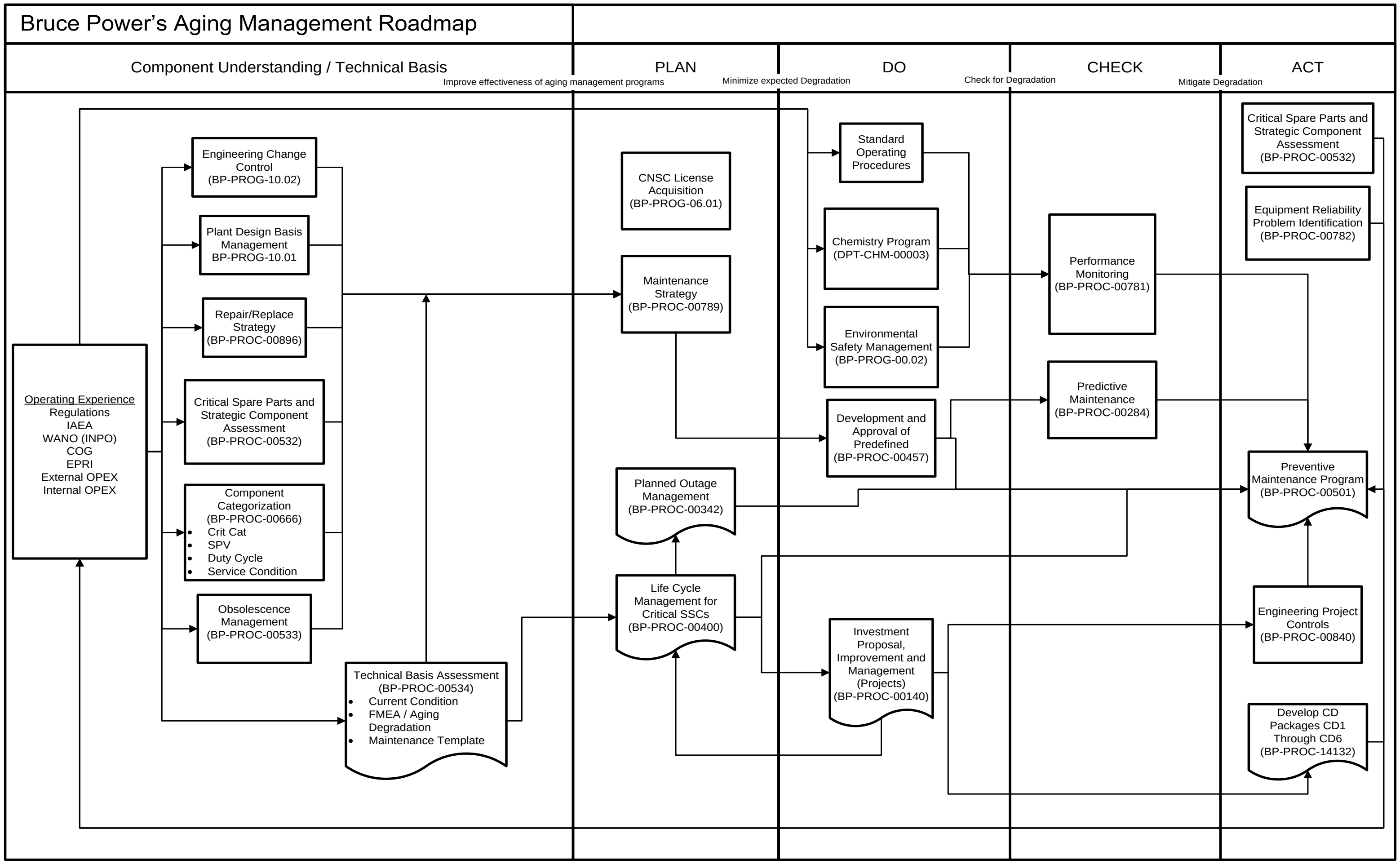

Figure 15: Overview of Governance for Ageing

6.1.4.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 4 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-4 with those under 'Plant' are discussed. Programs, processes and documentation under SF-1 and SF-3 provide the requirements and the bases for activities in ensuring actual condition and ageing impacts of SSCs important to safety are within the design basis. Outputs of the activities associated with SF-2 and SF-4 inform and provide feedback to each other. Results of the activities associated with SF-4 are fed back to those programs and processes under SF-1 and SF-3 for any follow-up and improvement opportunities. This relationship is illustrated in Figure 16.

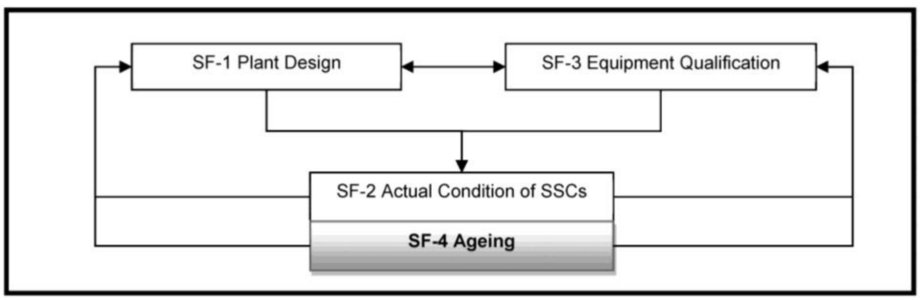

Figure 16: Safety Factor 4 Interfaces

SF-4 Ageing is very tightly coupled with SF-2 and SF-3 as it drives the scope of activities to maintain condition of SSCs important to safety within their design and operating envelope. Collectively SF-2, 3 and 4 provide input for design basis management and associated nuclear safety assessments to confirm acceptability of design and safety analysis margins.

6.1.4.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.1.4.2 are included in Section 5 of SFR 4 [17] [20].

One strength that was identified for both Bruce A and Bruce B during this review is as follows:

- Information from the Asset Management Program is proactively used to inform the business of the future needs related to ageing and to ensure the funding and priorities can be proactively established as required to ensure effective ageing management and plant safety.

The key issues (or, macro-gaps) arising from SFR 4 are provided verbatim in Table 4 and Table 5. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate, practicable opportunities for improvement are included in the IIP.

These reviews concluded that overall, ageing management at Bruce Power meets the requirements of the Safety Factor related to ageing with the exceptions noted in Table 4 and Table 5. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The review indicates that the current and planned implementations of the programs related to ageing are adequate to support continued safe and reliable operation of Bruce A and B.

Table 4: Key Issues Identified for SFR 4 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF4-1	NK21-PIP-20000-00001, "CSA N291 In-Service Inspection Program for Bruce NGS A Safety Related Structures" does not describe inspection requirements following an abnormal/environmental condition. Consideration should be given to revising NK21-PIP-20000-00001 to include inspection requirements following an abnormal/environmental condition.	Section 5.10 Micro-gaps against requirement clauses: CSA N291-08 – Clause 7.3.4
SF4-2	The specific requirements in CSA N285.4-14 on monitoring of fuel channel annulus spacer material properties will need to be addressed if Bruce Power is required to comply with this version of the standard in the future. Consideration should be given to developing guidance for monitoring annular spacer material properties.	Section 5.15 Micro-gaps against requirement clauses: CSA N285.4-14 – Clause 12.5

Table 5: Key Issues Identified for SFR 4 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF4-1	NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures [145] does not describe inspection requirements following an abnormal/environmental condition. Consideration should be given to revising NK29-PIP-20000-00001 to include inspection requirements following an abnormal/environmental condition.	Section 5.10 Micro-gaps against requirement clauses: CSA-N291-15 – Clause 7.3.4

Issue Number	Macro-Gap Description	Source(s)
SF4-2	The specific requirements in CSA-N285.4-14 on monitoring of fuel channel annulus spacer material properties will need to be addressed if Bruce Power is required to comply with this version of the standard in the future. Consideration should be given to developing guidance for monitoring annular spacer material properties.	Section 5.15 Micro-gaps against requirement clauses: CSA-N285.4-14 – Clause 12.5

6.2. Safety Analysis

This section summarizes the results of Safety Factors associated with safety analysis:

- SF-5 Deterministic Safety Analysis
- SF-6 Probabilistic Safety Analysis
- SF-7 Hazard Analysis

6.2.1. Deterministic Safety Analysis

6.2.1.1. Objective

The objective of the review of this Safety Factor is to determine to what extent the existing safety analysis remains valid when the following aspects have been taken into account:

- Actual plant design; the actual condition of SSCs and their predicted state at the end of the period covered by the PSR;
- Current deterministic methods; and current safety standards and knowledge.

In addition, the review should also identify any gaps relating to the application of the defence-in-depth concept.

6.2.1.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. Review of the application of analytical methods, guidelines and computer codes used in the existing deterministic safety analysis and comparison with current standards and requirements;
2. Review of the current state of the deterministic safety analysis (original analysis and updated analysis) for the completeness of the set of postulated initiating events forming the design basis, with consideration given to feedback of operating experience from plants of a similar design, in Canada;
3. Evaluation of whether the assumptions made in performing the deterministic safety analysis remain valid given the actual condition of the plant;
4. Evaluation of whether the actual operational conditions of the plant meet the acceptance criteria for the design basis;
5. Evaluation of whether the assumptions used in the deterministic safety analysis are in accordance with current regulations and standards;
6. Review of the application of the concept of defence-in-depth;

7. Evaluation of whether appropriate deterministic methods have been used for development and validation of emergency operating procedures and the accident management program at the plant;
8. Evaluation of whether calculated radiation doses and releases of radioactive material in normal and accident conditions meet regulatory requirements and expectations; and
9. Analysis of the functional adequacy and reliability of systems and components, the impact on safety of internal and external events, equipment failures and human errors, the adequacy and effectiveness of engineering and administrative measures to prevent and mitigate accidents.

6.2.1.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 5 [17] [20].

6.2.1.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the DSA processes. Deterministic Safety Analysis falls under the broader function of Nuclear Safety Assessment (NSA), which also covers activities such as probabilistic safety assessment and criticality safety assessment. The Nuclear Safety Assessment function, together with the Design Management Function, falls under Bruce Power's BP-PROG-10.01 Plant Design Basis Management. NSA and subsequently DSA are also initiated by various procedures under BP-PROG-11.01: Equipment Reliability, BP-PROG-10.02: Engineering Change Control and BP-PROG-11.04: Plant Maintenance via BP-PROG-10.01 and feeds back to these programs and procedures as illustrated in Figure 17.

DSA is executed systematically through a number of department level procedures in accordance with applicable regulatory documents, codes and standards defined in the PROL. Key implementing documents are listed in Section 4 of SFR 5 [17] [20] together with their brief description.

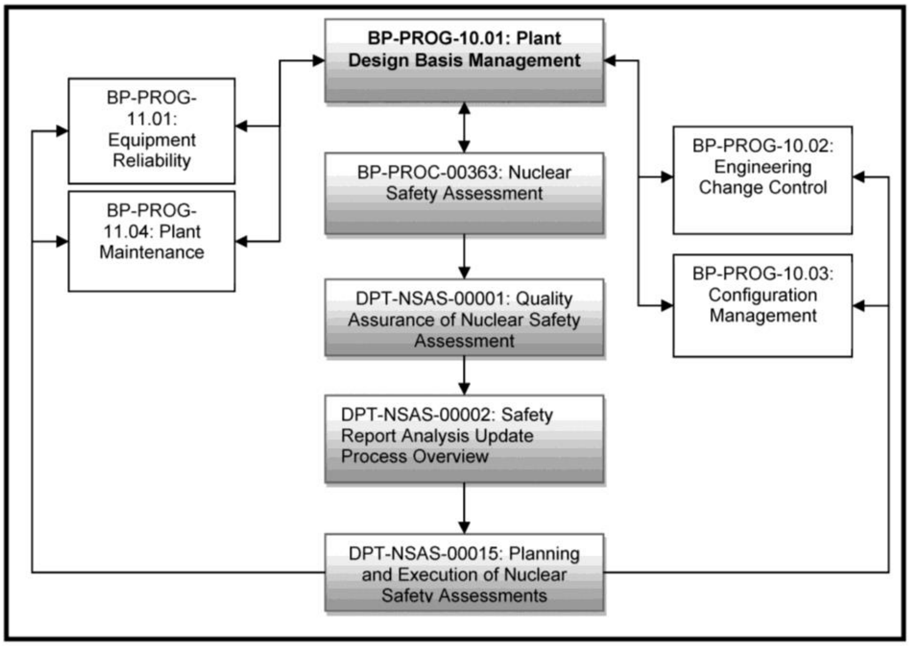

Figure 17: Overview of Governance for Safety Analysis

6.2.1.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 5 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-5 with those under 'Safety Analysis' are discussed. Safety Factor reviews under 'Safety Analysis' confirm that actual plant meets the deterministic and probabilistic safety analysis and design limits, as well as can withstand postulated hazards. Implementation of programs and procedures under Deterministic Safety Analysis utilize outputs of Hazard Analyses. Outputs of Deterministic Safety Analysis provide inputs for Probabilistic Safety Analysis. This relationship is illustrated in Figure 18.

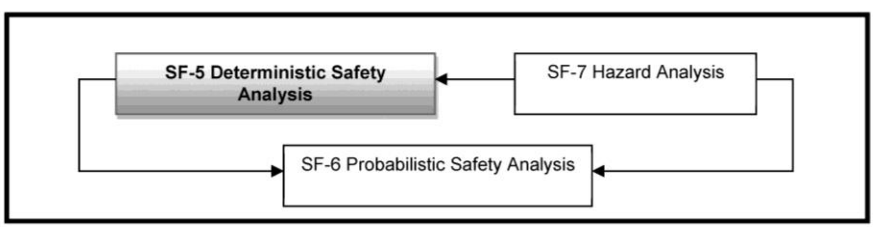

Figure 18: Safety Factor 5 Interfaces

6.2.1.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.2.1.2 are included in Section 5 of SFR 5 [17] [20].

Strengths relevant to DSA are as follows:

- Bruce Power has established an integrated strategy to improve the deterministic safety analysis contained in the Safety Reports as part of its objective to reach compliance with CNSC REGDOC-2.4.1 to the maximum practicable extent over a defined transition period. Bruce Power DSA procedures have been revised in consideration of CNSC REGDOC-2.4.1 requirements and the industry Principles and Guidelines for DSA (COG-09-9030 R01, Principles & Guidelines for Deterministic Safety Analysis Used in Licensing of Current CANDU Nuclear Power Plants Operating in Canada). Industry guidelines for Limit of the Operating Envelope (LOE)/Realistic Operating Envelope (ROE) and Best Estimate Analysis and Uncertainty (BEAU) methodologies are established. Moreover, “Derived Acceptance Criteria for Deterministic Safety Analysis” is issued as COG 13-9035. Bruce Power is leading or actively participating in all Safety Report Improvement (SRI) activities.
- Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support Severe Accident Management Guidance (SAMG) by mitigating severe accident progression and protecting containment integrity.

In addition, Bruce Power plans to perform supplementary evaluations of further improvements that could provide additional defence-in-depth. A project on improvements to the containment filtered venting system has been initiated to complete these additional evaluations, and is being tracked by AI 2015-07-3683.

The key issues (or macro-gaps) arising from SFR 5 are provided verbatim in Table 6 and Table 7. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate, practicable opportunities for improvement are included in the IIP.

In addition, the following acceptable deviations were identified:

- CNSC REGDOC-2.5.2 (Clause 4.3.3) – Bruce A and B
- CNSC G-144 – Bruce A and B
- CSA N290.1-13 (Clause 4.3.1.2) – Bruce A and B.

These reviews concluded that overall, Bruce Power meets the requirements of the Safety Factor related to Deterministic Safety Analysis and its conduct with the exceptions noted in Table 6 and Table 7. Practicable improvements to resolve the identified micro-gaps against the requirement clauses will enhance compliance to a level similar to those required for modern plants. The review indicates that the current and planned implementations of the programs related to

Deterministic Safety Analysis are adequate to support continued safe and reliable operation of Bruce A and B.

Table 6: Key Issues Identified for SFR 5 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF5-1	A number of the legacy analyses in the Safety Report are performed with codes (including the models and data) that have not been verified and validated to the requirements of CSA N286.7-99. Some key legacy computer codes, such as SOPHT, may not have been formally validated as per CSA N286.7-99, but code prediction has been compared to experimental and station data and benchmarking between SOPHT and TUF was performed. However, the following are not consistently addressed: • Assessment of the applicability of the codes to the analyzed events, and • Consideration of code accuracy in predicting key parameters.	Section 5.1 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 3 (Gap 2) REGDOC-2.4.1 – Clause 4.1 (Gap 1) REGDOC-2.4.1 – Clause 4.4.1 (Gaps 1, 2, 3 & 4) REGDOC-2.4.1 – Clause 4.4.2 (Gaps 1, 2, 3 & 5) REGDOC-2.4.1 – Clause 4.4.3 (Gaps 1, 2, 3 & 4) REGDOC-2.4.1 – Clause 4.4.5 (Gap 1) REGDOC-2.4.1 – Clause 4.5 (Gap 1) REGDOC-2.4.1 – Clause 4.7 (Gap 1) REGDOC-2.5.2 – Clause 9.4 (Gap 3) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.3.2 (Gap 2) REGDOC-2.4.1 – Clause 4.4.2.9 (Gap 1) REGDOC-2.4.1 – Clause 4.4.4.6 (Gap 1)
SF5-2	A systematic event identification and classification process is not well documented and/or demonstrated. AOOs have been addressed implicitly rather than explicitly in the deterministic safety analysis. Common-mode failure events are not included in Part 3 of the Safety Report. Relevant operational modes are not comprehensively addressed.	Section 5.2 and 5.8 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.3 (Gap 1) REGDOC-2.4.1 – Clause 4.2.1 (Gap 1) REGDOC-2.4.1 – Clause 4.2.2 (Gaps 1, 2, & 3) REGDOC-2.4.1 – Clause 4.2.3 (Gap 1) REGDOC-2.5.2 – Clause 4.2.1 (Gap 1) REGDOC-2.5.2 – Clause 4.2.3 (Gap 1) REGDOC-2.5.2 – Clause 6.1 (Gap 1) REGDOC-2.5.2 – Clause 6.4 (Gaps 1, 2, & 3) REGDOC-2.5.2 – Clause 6.6.1 (Gap 1) REGDOC-2.5.2 – Clause 7.4 (Gap 1) REGDOC-2.5.2 – Clause 8.10.4 (Gap 1) REGDOC-2.5.2 – Clause 9.1 (Gap 1) REGDOC-2.5.2 – Clause 9.2 (Gap 1) REGDOC-2.5.2 – Clause 9.4 (Gap 1) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.2.1 (Gaps 2 & 3)

Issue Number	Macro-Gap Description	Source(s)
SF5-3	The acceptance criteria are not systematically supported by experimental data.	Sections 5.4 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.3.2 (Gap 1) REGDOC-2.4.1 – Clause 4.3.4 (Gaps 1 & 2) REGDOC-2.5.2 – Clause 8.4.1 (Gap 1) REGDOC-2.5.2 – Clause 9.4 (Gap 2) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.3.2 (Gaps 3, 4, & 5) REGDOC-2.4.1 – Clause 4.3.4 (Gaps 3 & 4)
SF5-4	All analyses documented in the Safety Report were in accordance with the interpretation of the single failure criterion prevalent at the time. However, these analyses do not follow newer, more restrictive, interpretations of the single failure criterion.	Section 5.5 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.4.4 (Gaps 1, 2, & 4) REGDOC-2.5.2 – Clause 7.6.2 (Gap 1)
SF5-5	The limiting assumption with respect to RRS working or failed is not demonstrated for Small LOCA and transition breaks.	Section 5.5 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.4.4 (Gap 3)
SF5-6	Bruce A station PRA indicates that multi-unit events are considered. The completeness of such consideration needs to be confirmed, in particular, it may require complementary DSA for BDBAs, accounting for the capacity and limitations of long-term makeup water and electrical power supplies to confirm meeting the safety goals. Some of the analyzed events in the Safety Report will be classified as BDBAs and any required revision of their analysis will need to adopt a more realistic analysis methodology consistent with the PRA approach.	Section 5.7 Micro-gaps against requirement clauses: REGDOC-2.5.2 – Clause 4.2.2 (Gap 1) REGDOC-2.3.2 – Clause 3.4 (Gap 1) REGDOC-2.3.2 – Clause 4.2.1 (Gap 1) REGDOC-2.3.2 – Clause 4.2.5 (Gap 1) REGDOC 2.4.1 – Clause 4.4.4 (Gap 6)
SF5-7	It is not demonstrated if weather scenarios with probabilities of occurrences higher than 5% and dose calculations for intervals up to 1 year are considered.	Section 5.8 Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.4.4.7 (Gap 1)
SF5-8	Part 3 of the Safety Report is not fully reflective of the condition of the plant.	Section 5.3 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.6.2 (Gap 1)

Issue Number	Macro-Gap Description	Source(s)
SF5-9	Conservative assumptions are used in the analysis. However, there is no demonstration that the conservatism of the analysis covers modeling uncertainties.	Section 5.1 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.4.6 (Gap 1)
SF5-10	Cliff edge-effects are inherently covered in the assessment of trip coverage, however, it is not consistently addressed for quantitative acceptance criteria beyond reactor trip.	Section 5.1 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.2.3 (Gap 2) REGDOC-2.4.1 – Clause 4.4.2 (Gap 4)
SF5-11	Stress analysis for Bruce A shield cooling system is not performed to confirm the design and safety requirement.	Section 5.6 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 3 (Gap 1)
SF5-12	For accidents involving the irradiated fuel port, operator action is credited 10 minutes after the incident. This is less than the usual 15 minutes allowed from first unambiguous indication of a problem requiring operator action from inside the main control room.	Section 5.5 Micro-gaps against requirement clauses: REGDOC 2.4.1 – Clause 4.4.4 (Gap 5)

Table 7: Key Issues Identified for SFR 5 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF5-1	A number of the legacy analyses in the Safety Report are performed with codes (including the models and data) that have not been verified and validated to the requirements of CSA N286.7-99. The following are not consistently addressed: • Assessment of the applicability of the codes to the analyzed events, and • Consideration of code accuracy in predicting key parameters.	Section 5.1.4 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.1 (Gap 1) REGDOC-2.4.1 – Clause 4.4.1 (Gap 2) REGDOC-2.4.1 – Clause 4.4.2 (Gap 2) REGDOC-2.4.1 – Clause 4.4.5 (Gap 1) REGDOC-2.4.1 – Clause 4.5 (Gap 1) REGDOC-2.4.1 – Clause 4.7 (Gap 1) REGDOC-2.5.2 – Clause 9.4 (Gap 3) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.4.2.9 (Gap 1) REGDOC-2.4.1 – Clause 4.4.4.6 (Gap 1)
SF5-2	A systematic event identification and classification process is not well documented and/or demonstrated. AOOs have been addressed implicitly rather than explicitly in the deterministic safety analysis. Common-mode failure events are not included in Part 3 of the Safety Report. Relevant operational modes are not comprehensively addressed.	Section 5.2 and 5.8 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.2.1 (Gap 1) REGDOC-2.4.1 – Clause 4.2.2 (Gap 1) (Gap 2)

Issue Number	Macro-Gap Description	Source(s)
		REGDOC-2.4.1 – Clause 4.2.3 (Gap 1) (Gap 2) (Gap 4) (Gap 5) REGDOC-2.4.1 – Clause 4.3 (Gap 1) REGDOC-2.5.2 – Clause 4.2.1 (Gap 1) REGDOC-2.5.2 – Clause 4.2.3 (Gap 1) (Gap 2) REGDOC-2.5.2 – Clause 6.1 (Gap 1) REGDOC-2.5.2 – Clause 6.4 (Gap 1) (Gap 2) (Gap 3) REGDOC-2.5.2 – Clause 6.6.1 (Gap 1) REGDOC-2.5.2 – Clause 7.4 (Gap 1) (Gap 2) REGDOC-2.5.2 – Clause 9.1 (Gap 1) REGDOC-2.5.2 – Clause 9.2 (Gap 1) REGDOC-2.5.2 – Clause 9.4 (Gap 1) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.2.1 (Gap 2) (Gap 3) REGDOC-2.4.1 – Clause 4.2.2.5 (Gap 1) REGDOC-2.4.1 – Clause 4.3.2 (Gap 3) (Gap 5)
SF5-3	The quantitative acceptance criteria for AOOs and DBAs are not systematically supported by experimental data.	Section 5.4 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.3.2 (Gap 1) REGDOC-2.4.1 – Clause 4.3.4 (Gap 1) REGDOC-2.5.2 – Clause 8.4.1 (Gap 1) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.3.2 (Gap 4) REGDOC-2.4.1 – Clause 4.3.4 (Gap 3) (Gap 4)
SF5-4	All analyses documented in the Safety Report were in accordance with the interpretation of the single failure criterion prevalent at the time. However, these analyses do not follow the requirements of REGDOC-2.4.1 related to the single failure criterion.	Section 5.5 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.4.4 (Gap 1) REGDOC-2.5.2 – Clause 7.6.2 (Gap 1) Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.4.6 (Gap 2)
SF5-5	The limiting assumption with respect to RRS working or failed is not demonstrated for Small LOCA and transition breaks.	Section 5.5 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.4.1 (Gap 1) REGDOC-2.4.1 – Clause 4.4.4 (Gap 3)

Issue Number	Macro-Gap Description	Source(s)
SF5-6	It is not demonstrated if weather scenarios with probabilities of occurrences higher than 5% and dose calculations for intervals up to 1 year are considered.	Section 5.8 Micro-gaps against guidance clauses: REGDOC-2.4.1 – Clause 4.3.2 (Gap 2) REGDOC-2.4.1 – Clause 4.4.4.7 (Gap 1)
SF5-7	Conservative assumptions are used in the analysis. However, it has not been consistently demonstrated that the conservatism of the analysis covers modeling uncertainties.	Section 5.1.4 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.3.4 (Gap 2) REGDOC-2.4.1 – Clause 4.4.1 (Gap 3) REGDOC-2.4.1 – Clause 4.4.2 (Gap 5) REGDOC-2.4.1 – Clause 4.4.3 (Gap 1) REGDOC-2.4.1 – Clause 4.4.6 (Gap 1) REGDOC-2.5.2 – Clause 9.4 (Gap 2)
SF5-8	Cliff edge-effects are inherently covered in the assessment of trip coverage, however, it is not consistently addressed for quantitative acceptance criteria beyond reactor trip.	Section 5.1.4 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.2.3 (Gap 3) REGDOC-2.4.1 – Clause 4.4.2 (Gap 4)
SF5-9	For accidents involving the irradiated fuel port, operator action is credited 10 minutes after the incident. This is less than the usual 15 minutes allowed from first unambiguous indication of a problem requiring operator action from inside the main control room. For analysis of various loss of pressure control events, operator action is also credited at less than the usual 15 minutes allowed for operator action from inside the main control room.	Section 5.5 Micro-gaps against requirement clauses: REGDOC-2.5.2 – Clause 8.10.4 (Gap 1) CSA N290.1 – Clause 4.3.1.4 (Gap 1)
SF5-10	The Bruce B Safety Goals are less restrictive than the safety goals for new plants.	Section 5.9 Micro-gaps against requirement clauses: REGDOC-2.5.2 – Clause 4.2.2 (Gap 1)
SF5-11	A number of legacy analyses in the Bruce B Safety Report do not meet requirements of REGDOC-2.4.1 related to consequential failures, identification of important phenomena and initial and boundary conditions.	Section 5.1.4 Micro-gaps against requirement clauses: REGDOC-2.4.1 – Clause 4.4.2 (Gap 1) (Gap 3) REGDOC-2.4.1 – Clause 4.4.4 (Gap 2)

Issue Number	Macro-Gap Description	Source(s)
SF5-12	The Bruce B Safety Report does not include information to address the following requirements of CSA N288.2-14: Section 6.4.1.1 – Use of specialized models Section 6.5.1.1 – Justification of the chosen model for atmospheric dispersion.	Section 5.8 Micro-gaps against codes and standards: CSA N288.2-14 – Section 6.4.1.1 CSA N288.2-14 – Section 6.5.1.1

6.2.2. Probabilistic Safety Analysis

6.2.2.1. Objective

The objective of the review of this Safety Factor is to determine:

- The extent to which the existing Probabilistic Safety Assessment (PSA)[6] study remains valid as a representative model of the nuclear power plant;
- Whether the results of the PSA show that the risks are sufficiently low and well balanced for all postulated initiating events and operational states;
- Whether the scope (which should include all operational states and identified internal and external hazards), methodologies and extent (i.e., Level 1, 2 or 3) of the PSA are in accordance with current national and international standards and good practices; and
- Whether the existing scope and application of PSA are sufficient.

6.2.2.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. The existing PSA, including the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case;
2. Whether accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results;
3. Whether the scope and applications of the PSA are sufficient;
4. The status and validation of analytical methods and computer codes used in the PSA;

[6] Safety Factor Report 6 is titled “Probabilistic Safety Analysis”. However, Probabilistic Safety Analysis is referred to as Probabilistic Safety Assessment (PSA) throughout the document, with the exception of when it is specifically referring to Safety Factor Report 6; moreover, Probabilistic Risk Assessment (PRA) is equivalent to PSA.

5. Whether the results of PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria; and
6. Whether the existing scope and application of the PSA are sufficient for its use to assist the PSR global assessment, for example, to compare proposed improvement options.

6.2.2.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 5 [17] [20].

6.2.2.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Probabilistic Safety Assessment processes. Probabilistic Safety Assessment falls under the broader function of NSA, which also covers activities such as deterministic safety assessment and fitness for service assessments. The Nuclear Safety Assessment function, together with the Design Management Function, falls under Bruce Power's BP-PROG-10.01 Plant Design Basis Management. NSA and subsequently PSA is also initiated by various procedures under BP-PROG-11.01: Equipment Reliability, BP-PROG-10.02: Engineering Change Control, BP-PROG-11.04: Plant Maintenance and BP-PROG-12.01: Conduct of Plant Operations via BP-PROG-10.01 and feeds back to these programs and procedures as illustrated in Figure 19. This also demonstrates Bruce Power's use of risk informed decision making in many aspects of plant design and operation.

PSA is executed systematically through a number of department level procedures in accordance with applicable regulatory documents, codes and standards defined in the PROL[7]. Key implementing documents, captured in Figure 19, are listed in Section 4 of SFR 6 [17] [20] together with their brief description and also illustrated in Figure 19. It should be noted that the following PSA Guides also provide detailed guidance in the execution of PSA. They are not included in Figure 19 for simplicity.

- B-REP-03611-00005 for Level 1 At-Power
- B-REP-03611-00006 for Level 1 Outage
- B-REP-03611-00010 for Level 2 At-Power

In addition the following procedures under BP-PROG-10.01 Plant Design Basis Management also support implementation of PSA:

- BP-PROC-00335 Design Management

[7] Note, as of 2016, a project is underway to update Bruce Power governance for PSA to reflect changing regulatory requirements and model applications.

- BP-PROC-00582 Engineering Fundamentals
- BP-PROC-00502 Resolution of Differing Professional Opinions
- DIV-ENG-00009 Design Authority

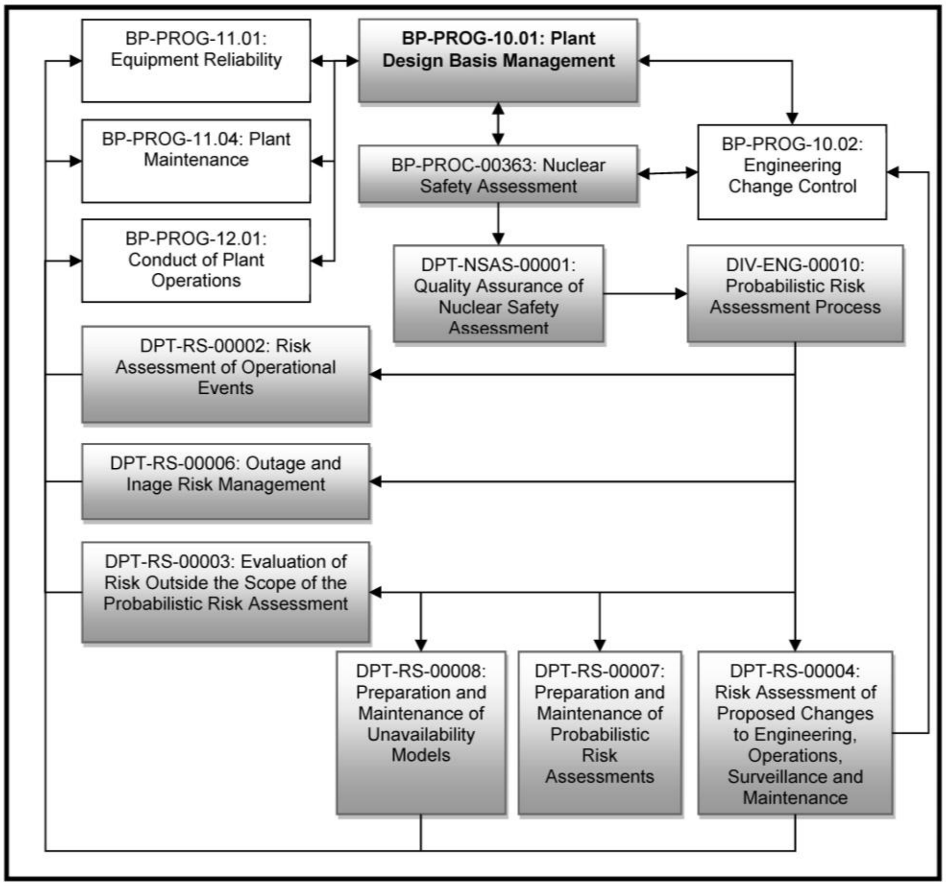

Figure 19: Overview of Governance for Probabilistic Safety Assessment

6.2.2.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 6 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-6 with those under 'Safety Analysis' are discussed. Safety Factor reviews under 'Safety Analysis' confirm that actual plant meets the deterministic and probabilistic safety analysis and design limits as well as can withstand postulated hazards. Probabilistic Safety Assessment integrates results of all postulated initiating events and combinations thereof. Hence, implementation of programs and procedures under Probabilistic Safety Assessment utilize outputs of both Hazard Analyses and Deterministic Safety Analysis. This relationship is illustrated in Figure 20.

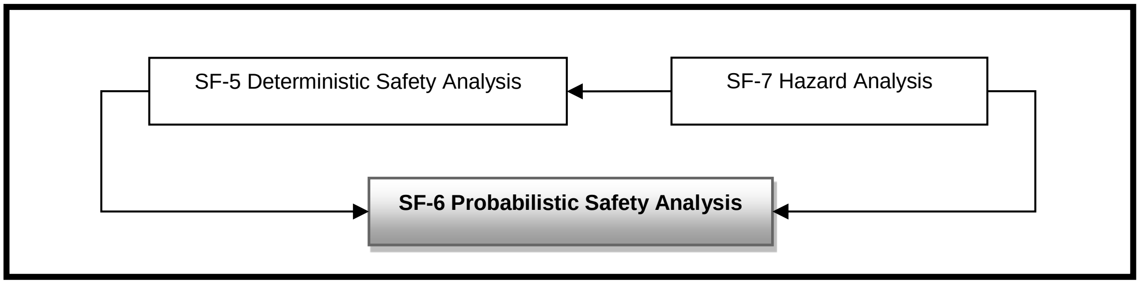

Figure 20: Safety Factor 6 Interfaces

6.2.2.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.2.2.2 are included in Section 5 of SFR 6 [17] [20].

One strength that was identified during this review for both Bruce A and Bruce B is as follows:

- Bruce Power has developed and implemented a process of continuous maintenance of the PRA model to ensure that the model is representative of the actual plant configuration and operation and testing at the station. This exceeds the requirement of CNSC REGDOC-2.4.2 (Clause 4.4) that the PRA models be updated every five years.

The key issues (or macro-gaps) arising from SFR 6 are provided verbatim in Table 8 and Table 9. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate, practicable opportunities for improvement are included in the IIP. In addition, the following acceptable deviations were identified:

- CNSC REGDOC-2.4.2 (Clause 4.3) – Bruce A and B
- CNSC REGDOC-2.5.2 (Clause 7.6) – Bruce A.

These reviews concluded that overall, Bruce Power meets the requirements of the Safety Factor related to Probabilistic Safety Assessment with the exceptions noted in Table 8 and Table 9. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The review indicates that the current and planned implementation of the programs related to Probabilistic Safety Assessment is adequate to support continued safe and reliable operation of Bruce A and B.

Table 8: Key Issues Identified for SFR 6 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF6-1	Although the result of each separate PRA meets the safety goal limits set up for Bruce A PRAs, their aggregates obtained by summation across all available PRA types, do not meet the more stringent quantitative safety goal targets set forth in the requirement clause.	Section 5.5.1 Micro-gaps against requirement clauses: REGDOC-2.5.2 – Clause 4.2.2
SF6-2	The proposed safety goal that the contribution to the large release frequency from all sequences involving failure to shut down be below 10^{-7}/yr events per reactor per year is not met.	Section 5.5.3 Micro-gaps against guidance clauses: REGDOC-2.5.2 – Clause 8.4.2

Table 9: Key Issues Identified for SFR 6 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF6-1	The aggregate SCDF and LRF obtained by summation across all available PRA types do not meet the safety goal targets set forth in the requirement clause 4.2.2 of CNSC REGDOC-2.5.2, although they meet the one order of magnitude higher limits defined by Bruce Power in Level 2 PRA guide B-REP-03611-00010 Rev 1.	Section 5.5.1 Micro-gaps against requirement clause: REGDOC-2.5.2 – Clause 4.2.2

6.2.3. Hazard Analysis

6.2.3.1. Objective

The objective of the review of this Safety Factor is to determine the adequacy of protection of the nuclear power plant against internal and external hazards with account taken of the actual plant design, actual site characteristics, the actual condition of SSCs and their predicted state at the end of the period covered by the PSR, and current analytical methods, safety standards and knowledge.

6.2.3.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. For each internal or external hazard identified, include the adequacy of the protection, with account taken of the following:
    a. The credible magnitude and associated frequency of occurrence of the hazard;
    b. Current safety standards;
    c. Current understanding of environmental effects;
    d. The capability of the plant to withstand the hazard as claimed in the safety case, based on its current condition and with allowance given to predicted ageing degradation;
    e. The appropriateness of procedures to cover operator actions claimed to prevent or mitigate the hazard.
2. Check list of internal and external hazards for completeness.
    a. The following is a representative list of internal hazards that may affect plant safety:
        i. Fire (including measures for prevention, detection and suppression of fire);
        ii. Flooding;
        iii. Pipe whip;
        iv. Missiles and drops of heavy loads;
        v. Steam release;
        vi. Hot gas release;
        vii. Cold gas release;
        viii. Deluge and spray;
        ix. Explosion;
        x. Electromagnetic or radio frequency interference;
        xi. Toxic and/or corrosive liquids and gases;
        xii. Vibration;
        xiii. Subsidence;
        xiv. High humidity;
        xv. Structural collapse;
        xvi. Loss of internal and external services (cooling water, electricity, etc.);
        xvii. High voltage transients; and
        xviii. Loss or low capacity of air conditioning (which may lead to high temperatures).
    b. The following is a list of representative external hazards that may affect plant safety:
        i. Floods, including tsunamis;

ii. High winds, including tornadoes;
iii. Fire;
iv. Meteorological hazards (extreme temperatures, extreme weather conditions, high humidity, drought, snow, buildup of ice);
v. Sun storm;
vi. Toxic and/or corrosive liquids and gases, other contamination in the air intake (for example, industrial contaminants, volcanic ash);
vii. Hydrogeological and hydrological hazards (extreme groundwater levels, seiches);
viii. Seismic hazards;
ix. Volcano hazards;
x. Aircraft crashes, external missiles;
xi. Explosion;
xii. Biological fouling;
xiii. Lightning strike;
xiv. Electromagnetic or radio frequency interference;
xv. Vibration;
xvi. Traffic; and
xvii. Loss of internal and external services (cooling water, electricity, etc.).

6.2.3.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 7 [17] [20] [21].

6.2.3.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Hazard Analysis processes. The objective of hazard analysis is to determine the adequacy of protection of the nuclear power plant against internal and external hazards, with account taken of the actual plant design, actual site characteristics, and actual plant condition. As such, hazard analysis has both design verification and safety analysis aspects.

The programmatic guidance related to the design verification aspects of hazard analysis are covered under a set of procedures relating to specific hazards such as seismic events, fire and environmental qualification under BP-PROG-10.01 Plant Design Basis Management.

Safety analysis of hazards is covered under BP-PROC-00363 Nuclear Safety Assessment, which governs activities such as deterministic and probabilistic safety analyses under BP-PROG-10.01 Plant Design Basis Management. The programmatic guidance for risk evaluation and hazard screening of any hazard are probabilistic safety assessment procedures.

The implementation of BP-PROC-00363 Nuclear Safety Assessment is supported by a variety of divisional and departmental procedures. Although there is no specific procedure addressing hazard analysis, many of the procedures have general applicability and support the hazard analysis process. The implementation of BP-PROC-00335 Design Management is also supported by a variety of divisional and departmental procedures. A number of these are relevant to hazard analysis, since they address design provisions for specific hazards. The list of Bruce Power policies, programs and procedures that are relevant to hazard analysis is provided in Section 4 of SFR 7 [17] [20] [21] and their relationship is illustrated in Figure 21.

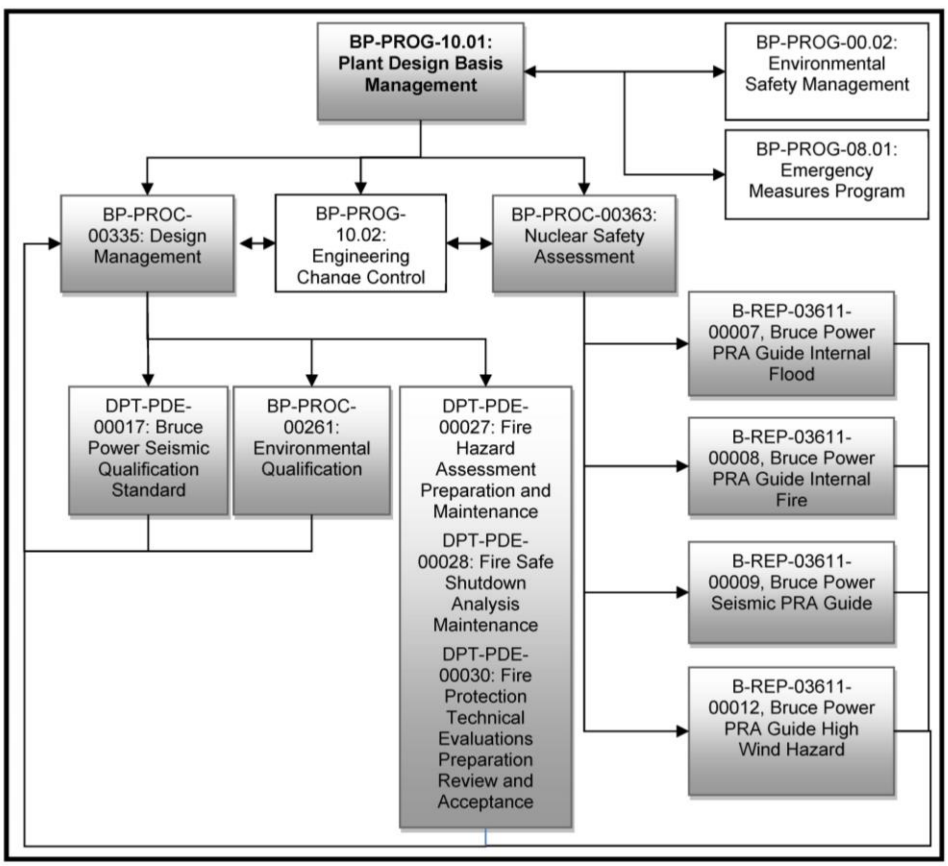

Figure 21: Overview of Governance for Hazard Analysis

6.2.3.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 7 [17] [20] [21].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-7 with those under 'Safety Analysis' are discussed. Safety Factor reviews under 'Safety Analysis' confirm that actual plant meets the deterministic and probabilistic safety analysis and design limits, as well as can withstand postulated hazards. Hazard Analysis demonstrates the ability of the plant to effectively respond to credible common-cause events and confirms that credited SSCs are qualified to survive and function during credible common-cause events. As such, outputs of Hazard Analysis are utilized both in deterministic and probabilistic safety analysis. This relationship is illustrated in Figure 22.

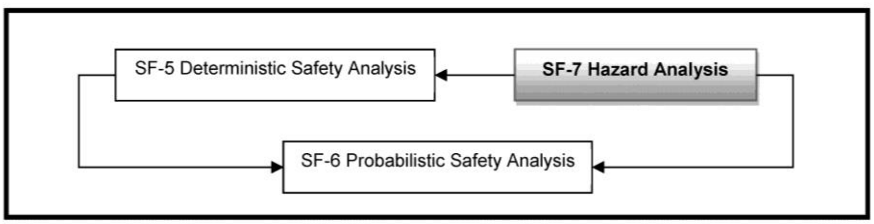

Figure 22: Safety Factor 7 Interfaces

6.2.3.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.2.3.2 are included in Section 5 of SFR 7 [17] [20] [21].

No specific strength was identified during this review.

The key issues (or macro-gaps) arising from SFR 7 for Bruce B are provided verbatim in Table 10. There were no key issues identified for Bruce A. Applicability of the Bruce B key issues to Bruce A is considered as part of the consolidation of the Safety Factor findings step during Global Assessment. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the "Source(s)" column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate, practicable opportunities for improvement are included in the IIP. In addition, the following acceptable deviation was identified:

- CNSC REGDOC-2.4.1 (Clause 4.2.1) – Bruce A.

These reviews concluded that overall, Bruce Power meets the requirements of the Safety Factor related to Hazard Analysis, with the exception noted in Table 10. Practicable improvements to resolve the identified micro-gap will enhance compliance to a level similar to those required for modern plants. The review indicates that the current and planned implementation of the programs related to Hazard Analysis is adequate to support continued safe and reliable operation of Bruce A and B.

Table 10: Key Issues Identified for SFR 7 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF7-1	Definition of DBE for purposes of seismic qualification of SSCs important to safety is not consistent with the 2013 version of the CSA standard.	Section 5.1.2 Micro-gaps against guidance clause: REGDOC-2.5.2 Clause 7.13.1

6.3. Performance and Feedback from Operating Experience

This section summarizes the results of Safety Factors associated with the physical plant:

- SF-8 Safety Performance
- SF-9 Use of Experience from Other Plants and Research Findings

6.3.1. Safety Performance

6.3.1.1. Objective

The specific objective of the review of this Safety Factor is to determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, indicate need for safety improvements.

6.3.1.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including:
    a. Safety related incidents, low level events and near misses;
    b. Safety related operational data;

c. Maintenance, inspection and testing;
d. Replacements of Structures, Systems and Components (SSCs) important to safety owing to failure or obsolescence;
e. Modifications, either temporary or permanent, to SSCs important to safety;
f. Unavailability of safety systems;
g. Radiation doses (to workers, including contractors);
h. Off-site contamination and radiation levels;
i. Discharges of radioactive effluents;
j. Generation of radioactive waste;
k. Compliance with regulatory requirements.

2. Where safety performance indicators are used, the review considers their adequacy and effectiveness, applying trend analysis and comparing performance levels with those for other plants in Canada;
3. The review considers the effectiveness of the processes and methodology used to evaluate and assess operating experience and trends. The findings of the reviews of other Safety Factors is taken into account when undertaking this task;
4. Records of radiation doses and radioactive effluents are reviewed to determine whether these are within prescribed limits, as low as reasonably achievable and adequately managed. Although radiation risks is considered in all Safety Factors, the review of this Safety Factor examines specifically data on radiation doses and radioactive effluents and the effectiveness of the radiation protection measures in place. The review takes into account the types of activity being undertaken at the plant, which may not be directly comparable with those at other nuclear power plants in Canada; and
5. Data on the generation of radioactive waste will be reviewed to determine whether operation of the plant is being optimized to minimize the quantities of waste being generated and accumulated, taking into account the national policy on radioactive discharges and international treaties, standards and criteria.

6.3.1.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 8 [17] [20].

6.3.1.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Safety Performance processes. BP-MSM-1 Bruce Power Management System Manual describes the vision, values, key result areas, policies, programs and procedures for achieving excellence in safety and performance in all aspect plant operations. It defines how Bruce Power executes plant operation, manages performance and assesses results to achieve continuous improvement. Central to this is fostering a healthy Safety Culture and being recognized for excellence in all aspects of nuclear safety including reactor safety, radiation safety, personnel safety and environmental safety management.

From a Safety Performance perspective the key implementing documents are those covering the availability of SSCs to perform their safety functions when called upon during an abnormal operational occurrence, a design basis event, design extension condition or beyond design basis event involve those covering the programmatic and process aspects of condition assessment and performance monitoring. During normal operation the more relevant programmatic and process aspects involve day-to-day monitoring, prevention, mitigation and accommodation of radiation doses to workers and the public and similarly control or containment of radioactive materials and radioactive effluents to the environment.

The prevention aspects are covered by ensuring operations stays within the envelope established by the design and licensing basis. Operation of the plant within the design and operating envelope in accordance with the current PROL, with a particular focus on nuclear safety; have been discussed in greater detail in Safety Factors 1 through 7.

The programs and processes key in achieving safety performance in accordance with the licensing and design basis are described in Section 4 of SFR 8 [17] [20]. These programs and processes are listed in Section 4 of SFR 8 [17] [20] and their relationships are illustrated at a high level in Figure 23.

It should be noted that the relationship of programs in Figure 23 are arranged as ‘enabling programs’ for ‘operational programs’ which all support CNSC interface management that covers activities and associated communication with the CNSC to demonstrate compliance with PROL. In this context, safety performance, being a general topic, is assured by meeting and exceeding the requirements of the PROL in all aspects of plant operation at all times.

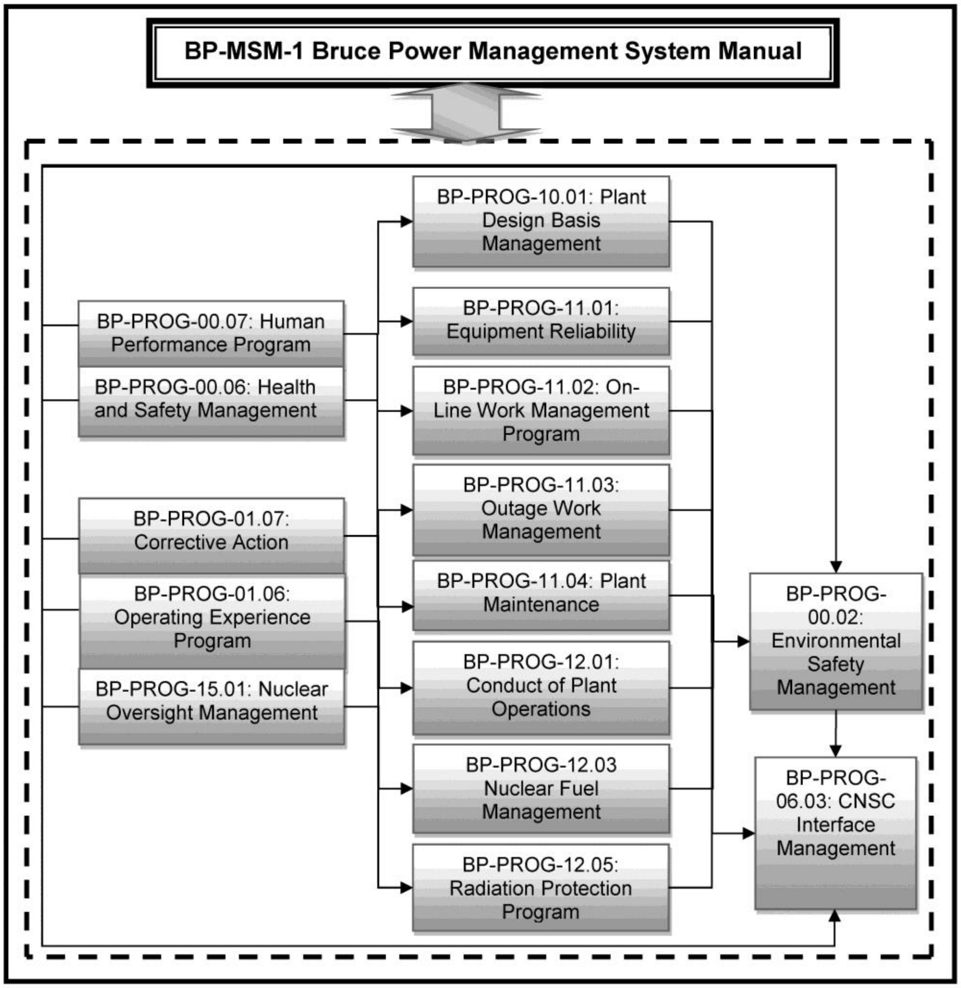

Figure 23: Overview of Governance for Safety Performance

6.3.1.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 8 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interface of SF-8 with SF-9 are discussed. As discussed in Section 6.3.1.4, SF-8 Safety Performance covers a number of ‘enabling’ and ‘operating ‘programs including Operating Experience Program which is one of the ‘enabling programs’ that support all ‘operating’ programs. All programs covered under SF-8 support compliance with the PROL and environmental legislation. Figure 24 demonstrates the importance of continuous safety performance improvement through sharing of OPEX with other plants to achieve excellence in nuclear safety.

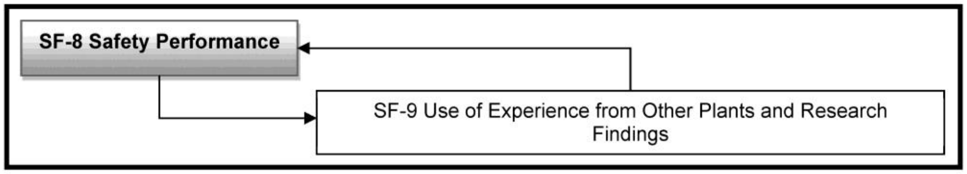

Figure 24: Safety Factor 8 Interfaces

6.3.1.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.3.1.2 are included in Section 5 of SFR 8 [17] [20].

An observed strength involves the commitments to improvements that are systematically being undertaken, based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently.

Nuclear Oversight and Regulatory Affairs (NORA) and Performance Improvement documents that summarize information for easier review by management include:

- Quarterly NORA Oversight Reviews covering audits and performance based assessments per BP-PROG-15.01, Nuclear Oversight Management; and
- Quarterly Focus Area Self Assessment Status & Summary Reports from Performance Improvement per BP-PROG-01.06, Operating Experience Program.

Furthermore, the audit organization has a well-developed Auditor Training program which used a Systematic Approach to Training based training design. Job Task Analysis is documented for knowledge and skill elements. The training program is documented and aligned to develop proficient auditors upon completion of qualifications. Auditors are professional and meet expectations of managers for performance as qualified auditors.

Bruce Power's organization shares Safety Performance OPEX, Compliance Reporting and Corrective Action processes as commonly-maintained programs between Bruce A and Bruce B, and thus observations and lessons learned can be used at both Bruce A and Bruce B. Additionally, there is an opportunity to share knowledge between the two stations by transferring managers from Bruce A to Bruce B and vice-versa. Thus, strengths at each station and means to see how the other Station prevents and mitigates less desirable situations are shared to increase the corporate knowledge and experience.

The key issues (or, macro-gaps) arising from SFR 8 are provided verbatim in Table 11 and Table 12. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the "Source(s)" column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement are included in the IIP.

These reviews concluded that overall, Bruce Power meets the requirements of the Safety Factor related to Safety Performance with the exceptions noted in Table 11 and Table 12. Practicable improvements to resolve the identified micro-gaps will enhance safety performance. The overall review indicates that the current and planned implementation of the programs related to Safety Performance is adequate to support continued safe and reliable operation of Bruce A and B

Table 11: Key Issues Identified for SFR 8 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF8-1	Governance procedures for the Integrated or Periodic Safety Review process need to be finalized to ensure staff understanding of the Regulatory direction.	Section 5.13
SF8-2	A risk-informed decision making process should be included in Equipment Reliability program so as to continually better prioritize activities.	Sections 5.14.2 and 7.3.1
SF8-3	The Safety Report improvement project needs to capture changes in Margin Management and adverse trend in the erosion of margin in LLOCA.	Section 5.3
SF8-4	The integrated time frame from conceptual design to station implementation for Nuclear Safety improvements that restore or improve margins (e.g., New Neutron Trip Project) needs to be reduced.	Section 5.7
SF8-5	Update the Safety Report Analysis of Record for single and dual Heat Transport pump events, with consideration of improvements, such as the modified 37-element fuel bundle.	Section 5.6
SF8-6	The documentation coverage for postulated initiating events not explicitly addressed in the Safety Report or PSAs needs to be improved. Neither the Safety Report deterministic safety analysis nor the PSAs explicitly include Crane Hazard analysis. Complete Hazard Analysis of Record and integrate it with the Deterministic Analysis and PSAs.	Section 5.7

Issue Number	Macro-Gap Description	Source(s)
SF8-7	Produce a document that explains the relationship and impact of the Fukushima type changes on the design basis, safety analyses and assessments, as they have been included in the licensing basis. This is necessary to ensure that the Design Basis and Configuration Management implications are understood. As appropriate, ensure Design Requirement and Design Manuals are updated appropriately, including capturing of Design Extension conditions if appropriate.	Section 5.13
SF8-8	Maintenance Backlogs were defined as needing improvement in the 2008 Bruce 3 and 4 ISR, based on a review of the backlog history. Although progress has been made on backlogs they are still identified as an area for improvement.	Sections 5.5, 7.3.1, and 7.3.2
SF8-9	Standby Class III Power System predicted unavailability targets exceeded in 2012 and 2013 due to an inconsistency between the modeling and plant operation. This requires correction [sic] action to reduce the unavailability.	Section 5.8
SF8-10	BP-PROC-00136 and BP-PROC-00169 are not affiliated with a Program.	Section 4.1, Table 4, Footnote 7

Table 12: Key Issues Identified for SFR 8 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF8-1	BP-PROC-00498 was to be revised, obsoleted or integrated in Equipment Reliability program so as to continually better prioritize activities.	Sections 5.14.2 and 7.3.1
SF8-2	The Safety Report Improvement Project needs to capture changes in Margin Management and adverse trend in the erosion of margin in LLOCA. The Safety Analysis Improvement Program needs to show the additional margins in LLOCA analysis by completing the work planned under the Composite Analysis Approach for LLOCA.	Section 5.3
SF8-3	The integrated time frame from conceptual design to station implementation for Nuclear Safety improvements that restore safety margins (e.g., heat transport high pressure trip on Units 3 and 4) should be reviewed to find	Section 5.7

Issue Number	Macro-Gap Description	Source(s)
	opportunities to more efficiently implement the safety improvement.	
SF8-4	The Safety Report Analysis of Record for single and dual Heat Transport pump events needs to be updated, with consideration of improvements, such as the modified 37-element fuel bundle.	Section 5.6
SF8-5	The documentation coverage for postulated initiating events not explicitly addressed in the Safety Report or Probabilistic Safety Assessments (PSAs) needs to be improved. Neither the Safety Report deterministic safety analysis nor the PSAs explicitly include Crane Hazard analysis. Complete Hazard Analysis of Record and integrate it with the Deterministic Analysis and PSAs Analysis of Records.	Section 5.7
SF8-6	Produce design documentation that explains the relationship and impact of the Fukushima type changes on the design basis, safety analyses and assessments, as they have been included in the licensing basis. This is necessary to ensure that when the Design Basis Assumptions change the changes to the Design Basis and Configuration Management implications are documented and understood. As appropriate, ensure Design Guides, Design Requirement and Design Manuals are updated appropriately, including capturing of Design Extension conditions if appropriate.	Section 5.13
SF8-7	Not all Bruce Power Programs readily map to the Safety Factor Reports. BP-PROC-01024 [4] should consider mapping each program to the respective Safety Factor Reports in Section 4.6 of the procedure to ensure completeness of items impacting the four pillars of safety. BP-PROC-00936 [104] should interface with BP-PROC-01024 [4] as the PSR is an input to the procedure.	Section 4.7
SF8-8	Updated versions of INPO documents are not always considered when governance documents are revised, nor was a governing procedure found to periodically review INPO, WANO and/or IAEA suggestions for improvement to confirm how they might improve Bruce Power governance documents.	Section 7.2
SF8-9	BP-PROC-00169 is not affiliated with a Program. Define the Program which BP-PROC-00169 implements.	Section 4.1, Table 6, footnote 6
SF8-10	The following PROGs, PROCs have not been revised within the required 3 year timeframe per BP-PROC-00166: General Procedure and Process Requirements and a review	Section 7.2

Issue Number	Macro-Gap Description	Source(s)
	of the PassPort action requests does not always provide evidence that the standard 3-year review has been completed and recommended no changes or whether the review has been deferred to a later date: BP-PROG-01.01-R005, Business Planning Program, February 5, 2010 [103] BP-PROG-11.02-R006, On-Line Work Management Program, October 2012 [159] BP-PROC-00169-R002, Safety Related System List, September 2007 [182] BP-PROC-00498-R006, Condition Assessment of Generating Units in Support of Life Extension, February 3, 2011 [144] BP-PROC-00735-R002, Long Range Cycle Planning Process, August 28, 2012 [162] BP-PROC-00795-R000, Human Performance Tools for Knowledge Workers, March 30, 2011 [102] BP-PROC-00839-R000, Reporting to CNSC/IAEA – Safeguards, June 21, 2012 [129] DPT-NSAS-00003-R004, Guidelines for Evaluating and Prioritizing Safety Report Issues, September 2011 [134] DPT-PE-00005-R000, Performance Requirements for Contamination Exhaust Control Filters, February 23, 2005 [148] SEC-EQD-00035-R002, Environmental Qualification Sustainability Monitoring, November 15, 2012 [131]	
SF8-11	ARs 28456029, 28456034, 28456045, on BP-PROC-00666, raised during AU-2014-00024 are not identified in PassPort against the document as either DCRs or ARs yet the audit ARs identity shortcomings against the document with respect to errors, omissions, misalignment and conflicting processes. Suggest all ARs against a document be linked to the document so users of the procedure are aware of the shortcomings.	PassPort and Section 7.2
SF8-12	Review of safety analysis to ensure it has been comprehensively been [sic] captured in the safe operating envelope via the Operational Safety Requirements documents.	Section 5.3

6.3.2. Use of Experience from Other Plants and Research Findings

6.3.2.1. Objective

The specific objective of the review of this Safety Factor is to determine whether there is adequate feedback of safety experience from nuclear power plants (both internal and external) or other pertinent operating experience from relevant non-nuclear facilities and of the findings of research.

6.3.2.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. Verify that arrangements are in place for the feedback of experience relevant to safety from other nuclear power plants and from relevant non-nuclear facilities;
2. Review the effectiveness of such programmes for the timely feedback of operating experience and for their output;
3. Review the processes for assessing and, if necessary, implementing research findings and findings from operating experience relevant to safety.

The emphasis of the tasks for Safety Factor 9 is on external experience. It is noted that Bruce Power operates two stations at the same site that are fundamentally of the same design, with some differences in SSCs due to in-service date of plants, with commensurate differences in their operation. In the context of SF-9 reviews, Bruce A and Bruce B are considered to be external plants to each other.

6.3.2.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 9 [17] [20].

6.3.2.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Use of Operating Experience from Other Plants and Research Findings processes. The main programs for feedback of safety experience from nuclear power plants (both internal and external) and of the findings of research are BP-PROG-01.06 Operating Experience Program and BP-PROG-01.07 Corrective Action. It should also be noted that these programs support all programs under BP-MSM-1 to enable continuous improvement of plant operations and safety culture and provide the basis for reporting of events to the CNSC in accordance with the PROL.

Bruce Power documents related to implementation of the elements related to the use and feedback of OPEX and Research and Development are listed in Section 4 of SFR 9 [17] [20] and their relationship is illustrated in Figure 25.

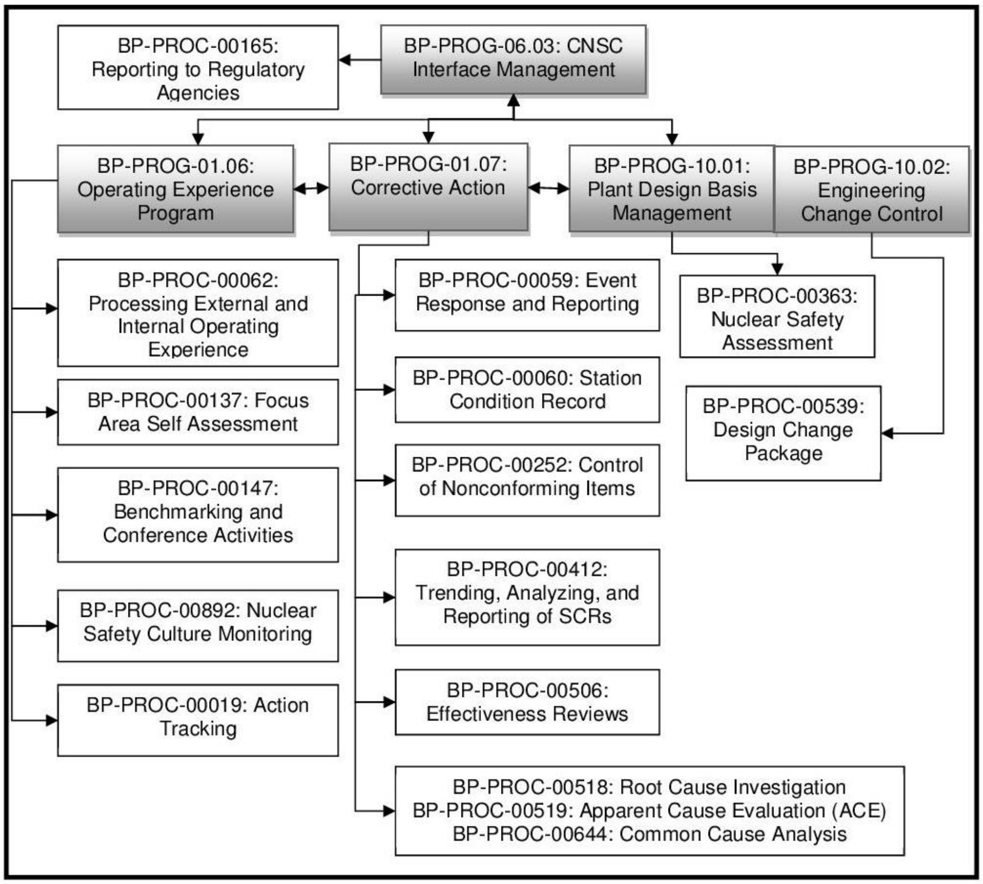

Figure 25: Overview of Governance for Use of Experience from Other Plants and Research Findings

6.3.2.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 9 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interface of SF-8 with SF-9 are discussed.

As discussed in Section 6.3.1.4, SF-8 Safety Performance covers a number of ‘enabling’ and ‘operating ‘programs including Operating Experience Program which is one of the ‘enabling programs' that support all 'operating' programs. All programs covered under SF-8 support compliance with the PROL and environmental legislation. In this context SF-9 supports all programs associated with safety performance. Figure 26 demonstrates the importance of sharing of OPEX with other plants to continuous safety performance improvement to achieve excellence in nuclear safety.

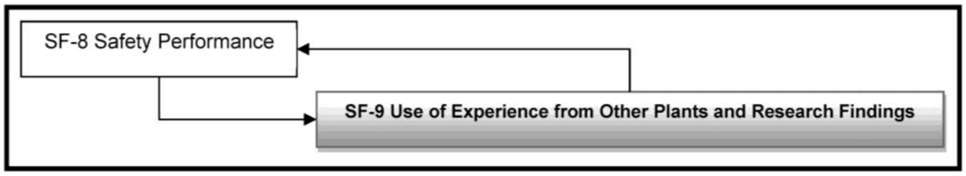

Figure 26: Safety Factor 9 Interfaces

6.3.2.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.3.2.2 are included in Section 5 of SFR 9 [17] [20].

The review demonstrates that Bruce Power's OPEX Program and its implementation provides for adequate feedback of safety experience from nuclear power plants (both internal and external) and of the findings of research in support of continued safe and reliable operation. In addition, the review demonstrates that Bruce Power does not confine itself to utilizing OPEX from nuclear power plants only, but makes use of OPEX from any industrial process plants. Moreover, research activities are being pursued and results are used to enhance nuclear safety and equipment performance and reliability. This is regarded as a strength in Bruce Power's OPEX Program.

The key issues (or, macro-gaps) arising from SFR 9 are provided verbatim Table 13 and Table 14. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the "Source(s)" column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement is included in the IIP.

These reviews concluded that overall, Bruce Power meets the requirements of the Safety Factor related to Use of Experience from Other Plants and Research Findings with the exception noted in Table 13. Practicable improvements to resolve the identified micro-gap will enhance the use of experience from other plants and research findings. The overall review indicates that the current and planned implementation of the programs related to Use of Experience from Other Plants and Research Findings is adequate to support safe and reliable continued operation of Bruce A and B.

Table 13: Key Issues Identified for SFR 9 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF9-1	Bruce Power participates widely in external conferences, symposia, research projects, but no specific governance was found that fosters this participation other than tangential references in BP-MSM-1 Sheet 2 and BP-PROG-09.02.	Section 5.3.1

It is noted that Section 8 of Bruce B SFR 9 states the following with respect to the gap in Table 13, and hence this gap was not considered further.

> *"Section 8 of the Bruce A Integrated Safety Review [17] included a gap SF9-1 indicating no specific governance could be found to ensure participation in external conferences, symposia, research projects. This has been addressed through a January 2016 revision of BP-PROG-01.06 [24] where wording from BP-PROC-00147 has been moved to Section 4.3, making it governance. In addition, Section 7.2.2.3 of this Safety Factor Report notes that 78 conferences and benchmarking activities had already been scheduled at the beginning of 2016 by the various functional area managers. This demonstrates that the responsible managers are performing the function."*

Table 14: Key Issues Identified for SFR 9 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF9-1	While the cerebral transport of knowledge is implicit in the stature and qualifications of the staff appointed to the CSA committees, governance surrounding their collection and use of OPEX in performing their duties in the various committees has not been found.	Section 5.3.1.2

6.4. Management

6.4.1. Organization and Administration

This section summarizes the results of Safety Factors associated with management:

- SF-10 Organization and Administration
- SF-11 Procedures
- SF-12 The Human Factor
- SF-13 Emergency Planning

6.4.1.1. Objective

The objective of the review of this Safety Factor is to determine whether the organization and administration are adequate for the safe operation of the nuclear power plant.

6.4.1.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. The review of the organization and management system will include a review of the following elements or programs against national and international standards:
    a. Policy statements of the operating organization;
    b. The documentation of the management system;
    c. The adequacy of arrangements for managing and retaining responsibility for activities or processes important to safety that have been outsourced (for example, maintenance and engineering services and safety analysis);
    d. The roles and responsibilities of individuals managing, performing and assessing work; and
    e. The processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved.
2. In addition, the review of the organization and management system will verify the following:
    a. There are adequate processes in place for managing organizational change;
    b. There is a human resource management process in place that ensures the availability of adequate, qualified human resources, including succession planning;
    c. There is adequate control of documents, products and records and this information is readily retrievable;
    d. There is adequate control of purchasing of equipment and services where this affects plant safety;
    e. There are adequate processes in place to check the quality of suppliers' management systems that are intended to ensure that equipment and services supplied to the nuclear power plant are fit for purpose and provided in an effective and efficient manner;
    f. There are adequate communication policies in place;
    g. There are adequate facilities for training and training programs are well structured;
    h. There are formal arrangements in place for employing suitably qualified internal and external technical, maintenance or other specialized staff;

   i. There are adequate processes in place for feedback of operating experience to the staff, including experience relating to organizational and management failures;

   j. There are suitable arrangements in place for maintaining the configuration of the nuclear power plant and operations are carried out in accordance with the safety analysis of the plant; and

   k. There are programs in place for ensuring continuous improvement, including self-assessment and independent assessment.

3. The review of the safety culture will include the following:

   a. A review of the safety policy to verify that it states that safety takes precedence over production and to confirm that this policy is effectively implemented;

   b. A review of procedures to ensure that nuclear and radiation safety are properly controlled and that appropriate measures are applied consistently and conscientiously by all staff;

   c. An assessment of the extent to which a questioning attitude exists and conservative decision making is undertaken in the organization;

   d. Verification that there is a strong drive to ensure that all events that may be instructive are reported and investigated to discover root causes and that timely feedback is provided to appropriate staff on findings and remedial actions;

   e. Verification that unsafe acts and conditions are identified and challenged in a constructive manner wherever and whenever they are encountered by plant employees and external staff (contractors);

   f. Verification that the organization has a learning culture and that it strives continuously for improvements and new ideas, and benchmarks against and searches out best practices and new technologies;

   g. Verification that there is an established and effective process for communication of safety issues;

   h. Verification that there is a process in place for prioritization of safety issues, with realistic objectives and timescales, that ensures that these issues receive proper resources;

   i. Verification that there is a method in place for achieving and maintaining clarity of the organizational structure and managing changes in accountability for matters affecting safety; and

   j. Verification that there is adequate training in safety culture, particularly for managers.

6.4.1.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 10 [17] [20].

6.4.1.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Organization and Administration processes. The management and operation of Bruce Power are defined by the programs and their implementing documents, as described in Bruce Power Management System (BPMS) Manual. The BP-MSM-1 provides a high level description of the way the business is managed, including the leadership direction defining how it is integrated. Nuclear safety is a primary consideration and the BPMS supports the enhancement and improvement of safety culture and the achievement of high levels of safety, as well as business performance, and is designed to ensure the leadership team can consistently deliver expected results and satisfy its stakeholders, such as the regulator, the public, its shareholders and employees. It ensures that Bruce Power meets the stipulations of its operating licences, other applicable codes, standards, legal and business requirements.

The BPMS covers six components, and applies to the entire business, at all locations managed by the organization. These components, which form the basis of the structure of this document, are:

- Strategic Direction.
- Plan - Policy, Program and Process Controls.
- Do - Process Management.
- Check - Monitoring for Results.
- Act - Continuous Learning
- Leadership and Organizational Accountability

The Management System Manual contains the company's vision, mission, values, behaviours, policies, key results areas, summary of the Board structure and a statement of commitment from the Chief Executive to the management system. It includes Sheets covering a summary of the complete list of Programs, a listing of Program owners and approvers, as well as functional area (process) groupings, the responsibilities and authorities of all section managers and above positions at Bruce Power and a summary of regulatory, legal and business requirements. Elements, or components, of the BPMS more frequently impacted by changes in the business or the external environment are captured in Sheets associated with BP-MSM-1 and can be revised without a revision to BP-MSM-1. These Sheets include:

- BP-MSM-1 SHEET 0001, MSM- Bruce Power Program Matrix – Sheet 0001

- BP-MSM-1 SHEET 0002, MSM - Approved Reference Chart Authorities and Responsibilities – Sheet 0002
- BP-MSM-1 SHEET 0003, MSM - List of Applicable Governing Acts, Regulations, Codes & Standards – Sheet 0003
- BP-MSM-1 SHEET 0004, MSM - Program Summaries – Sheet 0004

By design, the BPMS contributes to the establishment of a nuclear safety culture that assures reactor, environmental, industrial and radiological safety, during normal operations, as well as during extreme events. The BPMS serves as the overall quality assurance program, which complies with CSA N286, the standard required by the PROL.

As shown in Figure 27, full list of programs that govern all aspects of plant operations are given in BP-MSM-1 SHEET 0001, MSM- Bruce Power Program Matrix – Sheet 0001 in accordance with CSA N286-05. Figure 27 is arranged such that programs on the left column cover corporate and support functions whereas programs in the right column support operations. Grouping of functional areas, associated programs and the organizations responsible for their control are explicitly shown in Table 15.

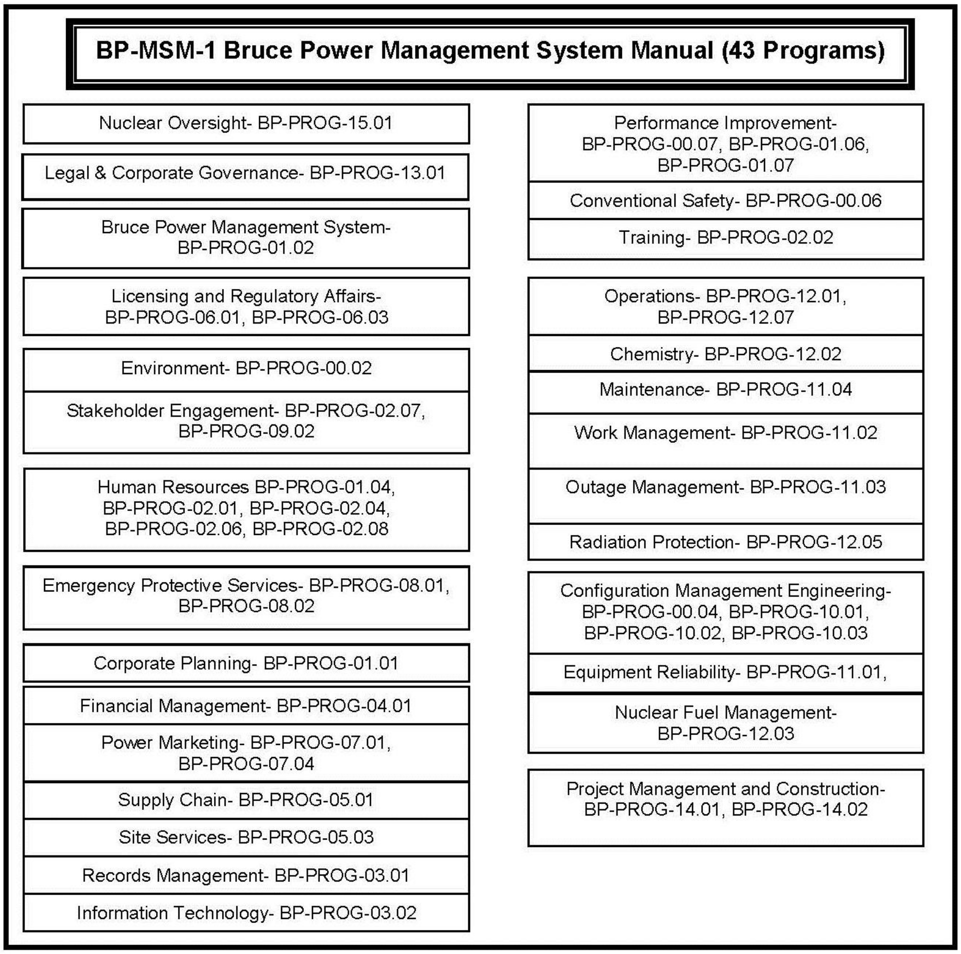

Figure 27: Overview of Governance for Organization and Administration

Table 15: Accountability Matrix for Functional Areas and Programs

Functional Area	Program Document	Program Name	Accountable Document Approver
Legal & Corporate Governance	BP-PROG-13.01	Corporate Governance and Legal Services	Chief Legal Officer & VP, Law & Emergency Management Division
Nuclear Oversight	BP-PROG-15.01	Nuclear Oversight Management	VP, Nuclear Oversight and Regulatory Affairs Division
BPMS	BP-PROG-01.02	Bruce Power Management System (BPMS) Management	VP, Nuclear Oversight and Regulatory Affairs Division
Licensing and Regulatory Affairs	BP-PROG-06.01	CNSC Licence Acquisition	VP, Nuclear Oversight and Regulatory Affairs Division

Functional Area	Program Document	Program Name	Accountable Document Approver
Licensing and Regulatory Affairs	BP-PROG-06.03	CNSC Interface Management	VP, Nuclear Oversight and Regulatory Affairs Division
Environment	BP-PROG-00.02	Environmental Safety Management	VP, Corporate Affairs Division
Stakeholder Engagement	BP-PROG-02.07	Employee Communications	VP, Corporate Affairs Division
Stakeholder Engagement	BP-PROG-09.02	Stakeholder Interaction	VP, Corporate Affairs Division
Human Resources	BP-PROG-01.04	Leadership Talent Management	EVP, Human Resources Group
Human Resources	BP-PROG-02.01	Worker Staffing	EVP, Human Resources Group
Human Resources	BP-PROG-02.04	Worker Development and Performance Management	EVP, Human Resources Group
Human Resources	BP-PROG-02.06	Worker/Labour Relations	EVP, Human Resources Group
Human Resources	BP-PROG-02.08	Total Rewards	EVP, Human Resources Group
Emergency Protective Services	BP-PROG-08.01	Emergency Measures Program	Chief Legal Officer & VP, Law & Emergency Management Division
Emergency Protective Services	BP-PROG-08.02	Nuclear Security	Chief Legal Officer & VP, Law & Emergency Management Division
Corporate Planning	BP-PROG-01.01	Business Plan Management	VP, Corporate Strategy and Business Development Division
Records Mgmt.	BP-PROG-03.01	Document Management	VP, Site Services Division
Information Technology	BP-PROG-03.02	Information Technology	VP & CIO, Information Technology Division
Financial Management	BP-PROG-04.01	Financial Reporting and Control	EVP, Finance and Commercial Services Group
Supply Chain	BP-PROG-05.01	Supply Chain	CFO & EVP, Finance & Commercial Services Group
Site Services	BP-PROG-05.03	Site Services	VP, Site Services Division
Power Marketing	BP-PROG-07.01	Electricity Revenue Management	VP, Commercial Services Division
Power Marketing	BP-PROG-07.04	Scheduling and Dispatch of Plant	VP, Commercial Services Division
Conventional Safety	BP-PROG-00.06	Health and Safety Management	VP, Nuclear Operations Support Division
Performance Improvement	BP-PROG-00.07	Human Performance Program	VP, Nuclear Operations Support Division
Performance Improvement	BP-PROG-01.06	Operating Experience Program	VP, Nuclear Operations Support Division

Functional Area	Program Document	Program Name	Accountable Document Approver
Performance Improvement	BP-PROG-01.07	Corrective Action	VP, Nuclear Operations Support Division
Training	BP-PROG-02.02	Worker Learning and Qualification	VP, Nuclear Operations Support Division
Work Management	BP-PROG-11.02	On-line Work Management Program	VP, Nuclear Operations Support Division
Outage Mgmt.	BP-PROG-11.03	Outage Work Management	VP, Nuclear Operations Support Division
Maintenance	BP-PROG-11.04	Plant Maintenance	VP, Nuclear Operations Support Division
Operations	BP-PROG-12.01	Conduct of Plant Operations	VP, Nuclear Operations Support Division
Chemistry	BP-PROG-12.02	Chemistry Management	VP, Nuclear Operations Support Division
Radiation Protection	BP-PROG-12.05	Radiation Protection Program	VP, Nuclear Operations Support Division
Operations	BP-PROG-12.07	Heavy Water Management	VP, Nuclear Operations Support Division
Configuration Management Engineering	BP-PROG-00.04	Pressure Boundary Quality Assurance Program	Chief Engineer & SVP, Engineering Division
Configuration Management Engineering	BP-PROG-10.01	Plant Design Basis Management	Chief Engineer & SVP, Engineering Division
Configuration Management Engineering	BP-PROG-10.02	Engineering Change Control	Chief Engineer & SVP, Engineering Division
Configuration Management Engineering	BP-PROG-10.03	Configuration Management	Chief Engineer & SVP, Engineering Division
Equipment Reliability	BP-PROG-11.01	Equipment Reliability	Chief Engineer & SVP, Engineering Division
Nuclear Fuel Management	BP-PROG-12.03	Nuclear Fuel Management	Chief Engineer & SVP, Engineering Division
Project Management and Construction	BP-PROG-14.01	Project Management and Construction	VP, Project Management & Construction Division
Project Management and Construction	BP-PROG-14.02	Contractor Management	VP, Project Management & Construction Division

6.4.1.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 10 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-10 with those under 'Management' are discussed.

Organization and Administration includes all of the policies and programs in BP-MSM-1 Bruce Power Management System Manual for safe and reliable operation of Bruce A and Bruce B in accordance with the PROL. Specifically, those programs located in the left column of Figure 27 pertain to the overall organization and management of the business in support of plant operation. In this context, SF-11 Procedures, SF-12 Human Factors and SF-13 Emergency Preparedness are specific elements of SF-10 Organization and Administration, i.e., the BP-MSM-1 Bruce Power Management System Manual. As such, their implementation informs the continuous improvement of organization and administration as illustrated in Figure 28 by the dashed lines.

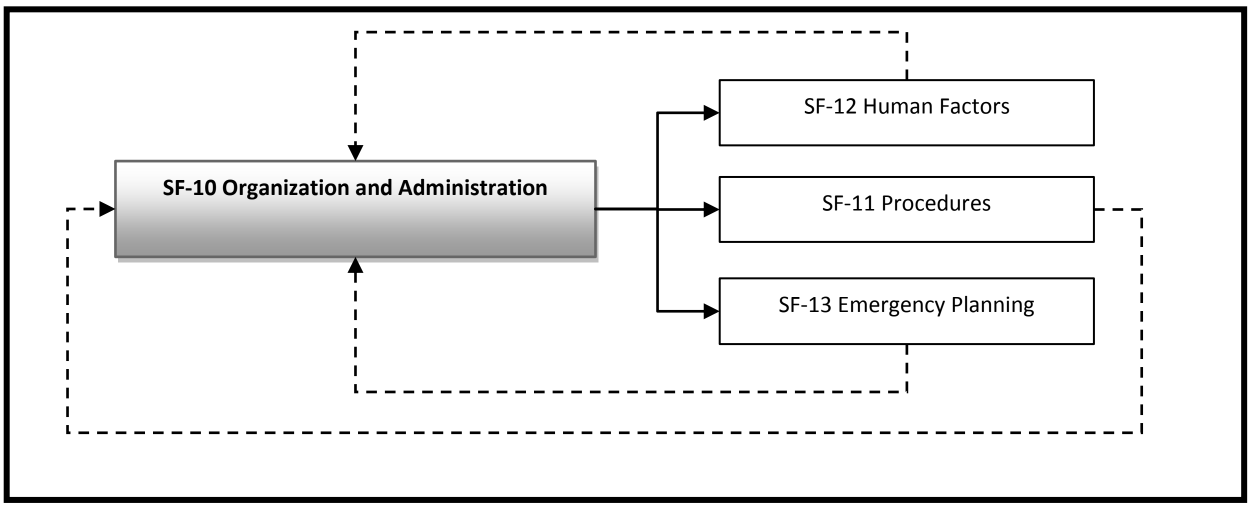

Figure 28: Safety Factor 10 Interfaces

6.4.1.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.4.1.2 are included in Section 5 of SFR 10 [17] [20].

Strengths identified during this review are:

- The existence of a comprehensive suite of programs and procedures that ensure the organization and administration will be controlled and well-documented in the future.

Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes.

- The commitments to improvements that are systematically being undertaken based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently.
- Bruce Power's organization shares Safety Performance OPEX, Compliance Reporting and Corrective Action processes, as commonly-maintained programs between Bruce A and Bruce B, so observations and lessons learned at Bruce B can be used at Bruce A and vice-versa. Additionally, there is an opportunity to share knowledge from Bruce A by transferring managers to Bruce B and vice-versa. Thus, strengths at each station and means to see how the other Station prevents and mitigates less desirable situations are shared to increase the corporate knowledge and experience.

The key issues (or, macro-gaps) arising from SFR 10 are provided verbatim in Table 16 and Table 17. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the "Source(s)" column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement is included in the IIP.

These reviews concluded that Bruce Power meets the requirements of the Safety Factor related to Organization and Administration with the exceptions noted in Table 16 and Table 17. Practicable improvements to resolve the identified micro-gaps will enhance Organization and Administration. The overall review indicates that the current and planned implementation of the programs related to Organization and Administration is adequate to support continued safe and reliable operation of Bruce A and B.

Table 16: Key Issues Identified for SFR 10 – Bruce A

Issue Number	Macro-Gap Description	Sources
SF10-1	Work Management Program BP-PROG-11.03 should be improved to address recurring outage issues identified through audits and FASAs.	Sections 5.2.5 and 7.2.1.6
SF10-2	BP-PROC-00363, Nuclear Safety Assessment, and its implementing documents should be revised to provide guidance on the responsibility of staff for Safety Assessment work performed outside of the NSAS Department.	Section 5.2.3

Issue Number	Macro-Gap Description	Sources
SF10-3	DCRs can become stagnant in the system, for example, depending on how they are initiated.	Section 5.3.3
SF10-4	BP-PROC-00136 is not affiliated with a Program.	Section 4.1, Table 4, footnote 6

Table 17: Key Issues Identified for SFR 10 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF10-1	BP-PROC-00363 [97], Nuclear Safety Assessment, and its implementing documents do not provide guidance on the responsibility of staff for Safety Assessment work performed outside of the NSAS Department.	Section 5.2.3
SF10-2	Ineffective implementation of BP-PROC-00060 [69]. DCRs can become stagnant in the system, for example, depending on how they are initiated which leads to documents being revised without incorporating the identified changes.	Section 5.3.3
SF10-3	A number of governance documents contain out of date references (e.g., superseded CNSC documents).	Section 5.3.3

6.4.2. Procedures

6.4.2.1. Objective

The objective of the review of this Safety Factor is to determine whether the operating organization’s processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.

6.4.2.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that review will examine a selection of the following procedures:

1. Operating procedures for normal and abnormal conditions (including anticipated operational occurrences, design basis accident conditions and post-accident conditions);

2. Procedures for the management of design extension conditions, including accidents with significant core degradation (for example, symptom based emergency operating procedures);
3. Maintenance, testing and inspection procedures;
4. Procedures for issuing work permits;
5. Procedures for controlling modifications to the plant design, procedures and hardware, including the updating of documentation;
6. Procedures for controlling the operating configuration;
7. Procedures for radiation protection, including procedures for on-site transport of radioactive material; and
8. Procedures for management of radioactive effluents and waste.

6.4.2.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 11 [17] [20].

6.4.2.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce Power programs and key documents for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety are listed in Section 4 of SFR 11 [17] [20]. The relationships amongst these programs are illustrated in Figure 29. As shown in Figure 29, these programs, within the context of BP-PROG-15.01 Nuclear Oversight Management, drive and feed back to each other.

Programs identified in Figure 29 are supported by detailed procedures which are also listed in Section 4 of SFR 11 [17] [20]. All programs shown in Figure 29 collectively ensure compliance with associated requirements of the PROL, as well as BP-OPP-00002: Operating Policies and Principles – Bruce A and BP-OPP-00001: Operating Policies and Principles – Bruce B.

Programs related to plant operation and supporting technical programs are shown in two adjacent blocks. These are the implementing programs that maintain compliance with operational limits and conditions and regulatory requirements. Corporate level programs for management system management, document management and corrective action ensure updating, maintenance and continuous improvement of all relevant programs and associated procedures. Adequacy and effectiveness of these programs are assured by the Nuclear Oversight Management Program.

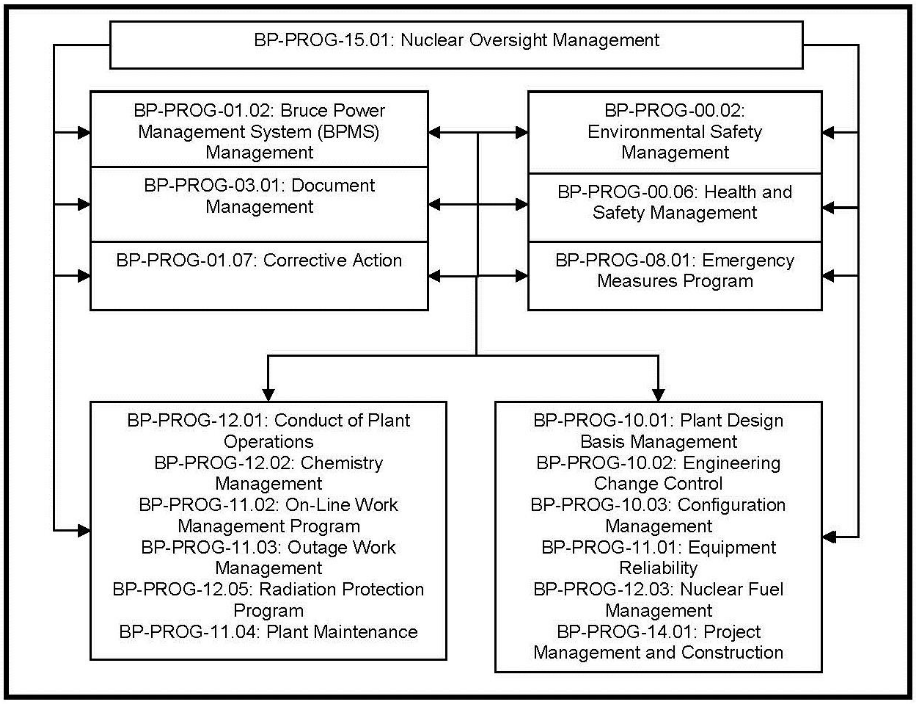

Figure 29: Overview of Governance for Procedures

6.4.2.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 11 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-11 with those under 'Management' are discussed.

Organization and Administration includes all of the policies and programs in BP-MSM-1 Bruce Power Management System Manual for safe and reliable operation of Bruce A and Bruce B in accordance with the PROL. Specifically, those programs located in the left column of Figure 27 pertain to the overall organization and management of the business in support of plant operation. In this context, SF-11 Procedures is a specific element of SF-10 Organization and Administration, i.e., BP-MSM-1 Bruce Power Management System Manual. As such, its implementation informs the continuous improvement of organization and administration as illustrated in Figure 30. In addition, implementation of programs and procedures identified under SF-11 inform and support programs implemented under SF-12 Human Factors and SF-13 Emergency Preparedness as illustrated in Figure 30.

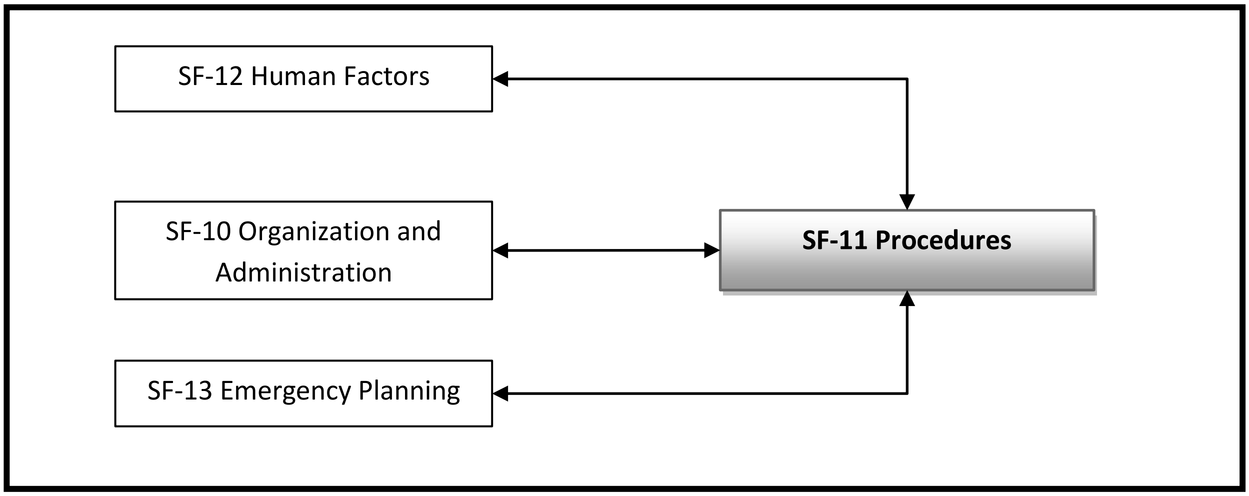

Figure 30: Safety Factor 11 Interfaces

6.4.2.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.4.2.2 are included in Section 5 of SFR 11 [17] [20].

Strengths identified during this review applicable to both Bruce A and Bruce B are:

- The existence of a comprehensive suite of programs and procedures that ensure procedures will be controlled and well documented in the future. Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes and revision of their procedures to meet best industry practice. This Safety Factor 11 review for Bruce A and Bruce B found that all aspects of the processes are satisfactory.
- The commitments to improvements that are systematically being undertaken based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently. These are discussed in detail in Safety Factor 10. This strength, however, is also directly applicable to the tasks identified for this Safety Factor and its assessment of procedures.

The key issues (or, macro-gaps) arising from SFR 11 are provided verbatim in Table 18 and Table 19. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement are included in the IIP.

These reviews concluded that Bruce Power meets the requirements of the Safety Factor related to Procedures with the exceptions noted in Table 18 and Table 19. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The overall review indicates that the current and planned implementation of the programs related to Procedures is adequate to support continued safe and reliable operation of Bruce A and B.

Table 18: Key Issues Identified for SFR 11 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF11-1	Difficulties in Maintenance Planning and Scheduling including meeting the expectations of On-Line Work Management Process and in performing maintenance in a timely manner are currently being experienced. High backlogs of PM Deferral Requests and PM Change Requests and the high number of multiple PM deferrals currently exist. Bruce Power had proactively identified PMOG as a management focus area and expected there should be positive changes to the overall program to bring the PM deferral numbers down and are actively addressing this gap. A graded approach is applied to backlogs to ensure safety significant backlogs are addressed in a timely manner.	Sections 5.4, 7.2.2.1, 7.3.1 and 7.3.2.
SF11-2	The selection of a radioactive waste processing method should include assessment of the maturity of technologies in relation to minimizing processing risks. This requirement is not explicitly identified in the Bruce Power procedures.	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 9.1
SF11-3	Dismantling and segmentation of equipment and/or structures should be considered to reduce radioactive waste volumes and to yield an improved packaging efficiency. This requirement is not explicitly identified in the Bruce Power procedures.	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 9.2.6
SF11-4	The concept of "storage for decay" is not identified in Bruce Power documentation.	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 11.2.1 CSA N292.3-14 – Clause 11.2.2 CSA N292.3-14 – Clause 11.2.3
SF11-5	BP-PROC-00498 on Condition Assessments is out of date and has been committed for future revision. The procedure needs to be updated or superseded by existing procedures which adequately capture the necessary information.	Section 5.4

Table 19: Key Issues Identified for SFR 11 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF11-1	The selection of a radioactive waste processing method should include assessment of the maturity of technologies in relation to minimizing processing risks. This requirement is not explicitly identified in the Bruce Power procedures.	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 9.1
SF11-2	Dismantling and segmentation of equipment and/or structures should be considered to reduce radioactive waste volumes and to yield an improved packaging efficiency. This requirement is not explicitly identified in the Bruce Power procedures.	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 9.2.6
SF11-3	The concept of “storage for decay” is not identified in Bruce Power documentation.	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 11.2.1 CSA N292.3-14 – Clause 11.2.2 CSA N292.3-14 – Clause 11.2.3
SF11-4	BP-PROC-00498 on Condition Assessments is out of date and has been committed for future revision. The procedure needs to be updated or superseded by existing procedures which adequately capture the necessary information.	Section 5.4
SF11-5	Bruce Power governance documents associated with Management of Radioactive Waste do not provide any information with respect to "treatments that mitigate exclusivity, pyrophoricity, and chemical reactivity".	Section 5.9 Micro-gaps against requirement clauses: CSA N292.3-14 – Clause 8.7

6.4.3. The Human Factor

6.4.3.1. Objective

The objective of the review of this Safety Factor is to determine the status of the various human factors that may affect the safe operation of the nuclear power plant.

6.4.3.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. The review of human factors (HF) will consider the procedures and processes in place at the nuclear power plant to ensure the following:
    a. Adequate staffing levels exist for operating the plant, with due recognition given to absences, shift working and restrictions on overtime;
    b. Qualified staff are available on duty at all times;
    c. Adequate programs are in place for initial training, refresher training and upgrading training, including the use of simulators;
    d. Operator actions needed for safe operation have been assessed to confirm that assumptions and claims made in safety analyses (for example, Probabilistic Safety Assessment (PSA), deterministic safety analysis and hazard analysis) are valid;
    e. Human factors in maintenance are assessed to promote error-free execution of work;
    f. Adequate competence requirements exist for operating, maintenance, technical and managerial staff;
    g. Staff selection methods (for example, testing for aptitudes, knowledge and skills) are systematic and validated;
    h. Appropriate fitness for duty guidelines exist relating to hours, types and patterns of work, good health and substance abuse;
    i. Policies exist for maintaining the know-how of staff and for ensuring adequate succession management in accordance with good practices; and
    j. Adequate facilities and programs are available for staff training.
2. The following aspects of the Human-Machine Interface (HMI) will be subjected to an overall review to determine if the HMI continues to be satisfactory:
    a. Design of the control room and other workstations relevant to safety;
    b. Human information requirements and workloads; and
    c. Clarity and achievability of procedures.

6.4.3.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 12 [17] [19] [20].

6.4.3.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Human Factor processes. BP-PROG-00.07: Human Performance Program describes Bruce Power's systematic approach to improving human performance through the use of event-free tools, managing defences, and other elements that enhance human performance. Bruce Power's Human Performance Program uses a strategic approach to managing Human Performance by reducing errors and managing defences. Bruce Power's Human Performance Program identifies four lines of defence or control to improve station resilience to human error and related events: Administrative Controls; Culture Controls; Oversight Control; and Engineered Controls. The implementation of these controls is discussed briefly.

Bruce Power implements Administrative Controls through the programs that govern the development of procedures, training, and work processes.

The lines of defence associated with Cultural and Oversight controls as defined in BP-PROG-00.07: Human Performance Program

DPT-PDE-00013 Human Factors Engineering Program Plan, invoked by BP-PROC-00335 Design Management under BP-PROG-10.01 Plant Design Basis Management, focuses on ensuring that Human Factors is considered in design and provides an input to the development of Engineered Controls through design as a line of defence. Engineered controls are embedded in BP-PROG-12.01: Conduct of Plant Operations and BP-PROG-11.04: Plant Maintenance.

Key implementing documents for human performance are listed in Section 4 of SFR 12 [17] [19] [20]. The relationships amongst these programs are illustrated in Figure 31.

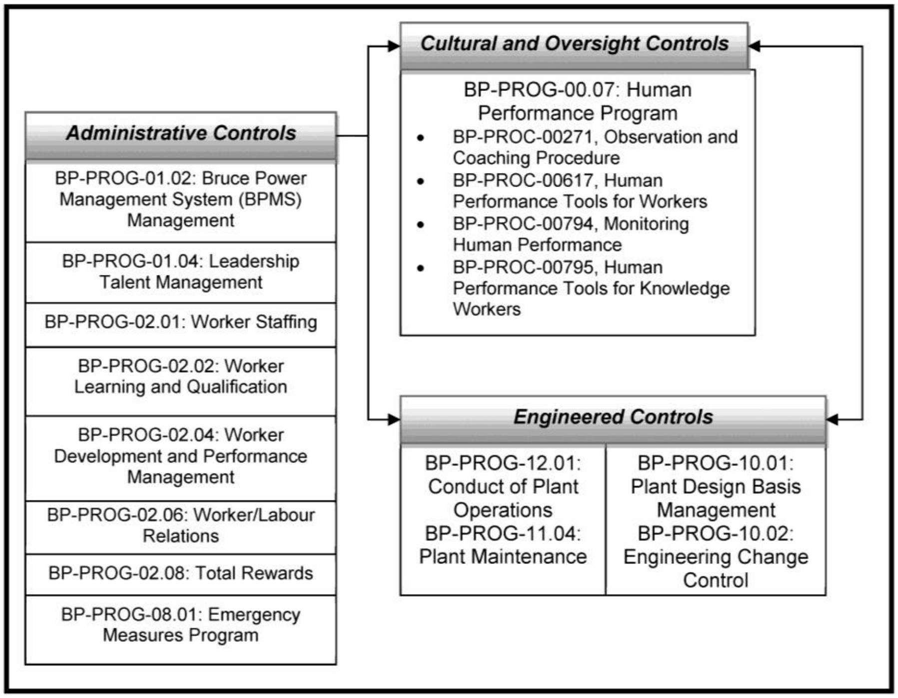

Figure 31: Overview of Governance for Human Factors

6.4.3.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 12 [17] [19] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-12 with those under 'Management' are discussed.

Organization and Administration includes all of the policies and programs in BP-MSM-1 Bruce Power Management System Manual for safe and reliable operation of Bruce A and Bruce B in accordance with the PROL. Specifically, those programs located in the left column of Figure 29 pertain to the overall organization and management of the business in support of plant operation. In this context, SF-12 Human Factors is a specific element of SF-10 Organization and Administration, i.e., BP-MSM-1 Bruce Power Management System Manual. As such, its implementation informs the continuous improvement of organization and administration as illustrated in Figure 32. In addition, implementation of programs and procedures identified under SF-12 inform and support programs implemented under SF-10 Organization and Administration and SF-13 Emergency Preparedness as illustrated in Figure 32.

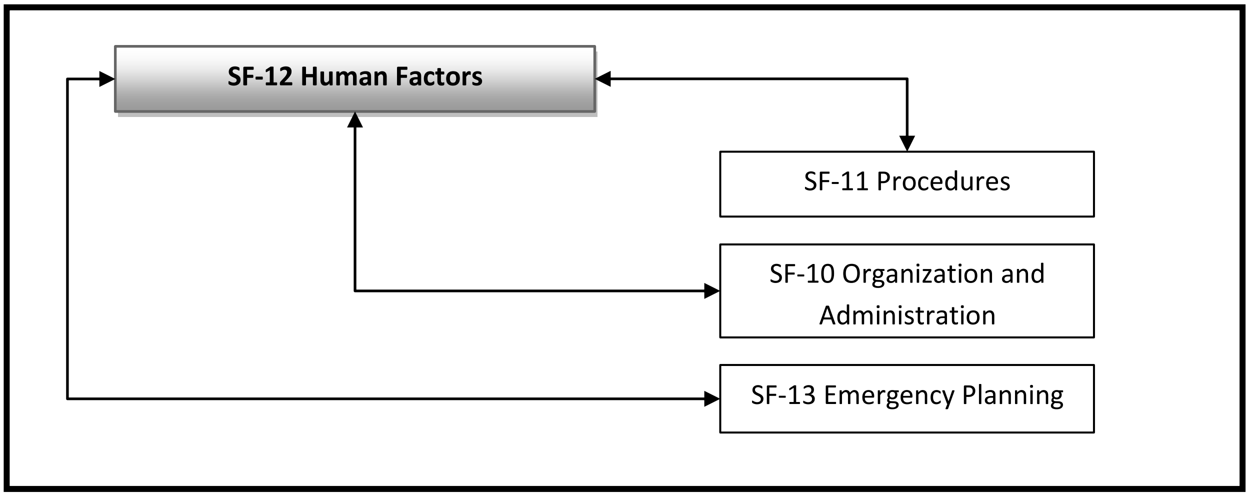

Figure 32: Safety Factor 12 Interfaces

6.4.3.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.4.3.2 are included in Section 5 of SFR 12 [17] [19] [20].

No specific strengths were related to Human Factors during this review.

The key issues (or, macro-gaps) arising from SFR 12 are provided verbatim in Table 20 and Table 21. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the "Source(s)" column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement are included in the IIP.

In addition, the following acceptable deviations were identified:

- CSA N290.12 (Clause 6.1.1) – Bruce B
- CSA N290.12 (Clause 6.1.5) – Bruce B.

These reviews concluded that Bruce Power meets the requirements of the Safety Factor related to Human Factor with the exceptions noted in Table 20 and Table 21. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The overall review indicates that the current and planned implementation of the programs related to Human Factors is adequate to support continued safe and reliable operation of Bruce A and B.

Table 20: Key Issues Identified for SFR 12 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF12-1	A review of Bruce Power documentation could not confirm that all operator actions under accident conditions have been assessed and confirmed valid. While it is clear that all credited human actions, as noted in the Bruce A PRA and included in AIMs were validated, it is not clear whether human actions identified in the Safety Report were a part of the credited human actions validated.	Section 5.4
SF12-2	The design of the control room and other workstations relevant to safety does not meet the guidance provided in NUREG-0700.	Section 5.11 Micro-gaps against guidance clauses: NUREG-0700 – Part 1 NUREG-0700 – Part 2

Table 21: Key Issues Identified for SFR 12 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF12-1	A review of internal self assessments on hours of work suggests that Bruce Power is maintaining staffing levels but not without violations that do not seem to be decreasing overall. Therefore, while programs for ensuring adequate staff levels are adequate, they are not being effectively implemented.	Sections 5.1 and 7.1.1
SF12-2	Lack of input from training exercises, particularly those modeling accident conditions, to safety analyses to validate assumptions.	Section 5.4
SF12-3	A review of Bruce Power documentation could not confirm that all operator actions under accident conditions have been assessed and confirmed valid. While it is clear that all credited human actions, as noted in the Bruce B Risk Assessment Report and included in AIMs were validated, it is not clear whether human actions identified in the Bruce B Safety Report were a part of the credited human actions validated.	Section 5.4
SF12-4	The design of the control room and other workstations relevant to safety may not meet some of the guidance provided in NUREG-0700.	Section 5.11 Micro-gaps against guidance clauses: NUREG-0700 – Part 1 NUREG-0700 – Part 2

Issue Number	Macro-Gap Description	Source(s)
SF12-5	Bruce Power Human Factors Engineering Program[8] does not meet some of the requirements and guidance in CSA N290.12.	Section 5.11 Micro-gaps against requirements clauses: CSA N290.12 – Clause 4.1.2 CSA N290.12 – Clause 4.1.6 CSA N290.12 – Clause 4.3 (Gap 1 and Gap 2) CSA N290.12 – Clause 6.3.1 CSA N290.12 – Clause 6.5.3 CSA N290.12 – Clause 6.5.4 CSA N290.12 – Clause 7.1 Micro-gaps against guidance clauses: CSA N290.12 – Clause 5.2.1 CSA N290.12 – Clause 5.2.2 CSA N290.12 – Clause 5.2.3 CSA N290.12 – Clause 5.2.4 CSA N290.12 – Clause 5.3.1 CSA N290.12 – Clause 5.3.2 CSA N290.12 – Clause 5.3.4 CSA N290.12 – Clause 5.4.2 CSA N290.12 – Clause 5.4.4 CSA N290.12 – Clause 5.5 CSA N290.12 – Clause 6.1.6 CSA N290.12 – Clause 6.2.2 CSA N290.12 – Clause 6.3.2 CSA N290.12 – Clause 6.3.3 CSA N290.12 – Clause 6.4.1 CSA N290.12 – Clause 6.4.2 CSA N290.12 – Clause 8.5 CSA N290.12 – Clause 8.6 CSA N290.12 – Clause 8.8 CSA N290.12 – Clause 8.9 CSA N290.12 – Clause 8.11 CSA N290.12 – Clause 8.12

[8] Note that this review was performed against DPT-PDE-00013-R008. This document was revised in June 2016 to ensure that it aligns with the requirements of CSA N290.12-14.

6.4.4. Emergency Planning

6.4.4.1. Objective

The objective of the review of this Safety Factor is to determine whether the operating organization has adequate plans, staff, facilities and equipment for dealing with emergencies at Bruce A and B and whether the operating organization's arrangements have been adequately coordinated with local and national systems and are regularly exercised.

6.4.4.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

1. An overall review will be performed to check that emergency planning at the plant continues to be satisfactory and to check that emergency plans are maintained in accordance with current safety analyses, accident mitigation studies and good practices.
2. It will be verified if the operating organization has given adequate consideration to significant changes at the site of the nuclear power plant and in its use, organizational changes at the plant, changes in the maintenance and storage of emergency equipment and developments around the site that could influence emergency planning.
3. Additionally :
    a. Evaluate the adequacy of on-site equipment and facilities for emergencies;
    b. Evaluate the adequacy of on-site technical and operational support centres;
    c. Evaluate the efficiency of communications in the event of an emergency, in particular the interaction with organizations outside the plant;
    d. Evaluate the content and efficiency of emergency training and exercises and check records of experience from such exercises;
    e. Evaluate arrangements for the regular review and updating of emergency plans and procedures;
    f. Examine changes in the maintenance and storage of emergency equipment; and
    g. Evaluate the effects of any recent residential and industrial developments around the site.

6.4.4.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 13 [17] [20].

6.4.4.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Emergency Planning processes. Bruce Power's BP-PROG-08.01 Emergency Measures Program (Level 1) defines the overall business need, constituent elements, functional requirements, implementing approaches and key responsibilities associated with the emergency management process. The objective of emergency measures is to develop and implement plans/procedures that mitigate or lessen the consequences of events that pose a hazard deemed unacceptable to staff, the public, the environment and/or the continuity of Bruce Power's business.

BP-PROG-08.01 Emergency Measures Program is implemented through six (6) Level 2 plans and one Level 2 procedure.

- BP-PLAN-00001 Bruce Power Nuclear Emergency Response Plan
- BP-PLAN-00002 Winter Storm Transportation Plan
- BP-PLAN-00003 Bruce Power Electricity Emergency Plan
- BP-PLAN-00004 Business Continuity Management
- BP-PLAN-00005 Radioactive Materials Transportation Emergency Response Plan
- BP-PLAN-00006 Conventional Emergency Management
- SEC-EPP-00007 Emergency Management Programs Assessment

In addition, BP-PROC-00010 Emergency Preparedness Drills and Exercises procedure describes the procedures for assessing emergency readiness. Continuous improvement to BP-PROG-08.01 Emergency Measures Program is achieved through BP-PROG-01.07: Corrective Action and BP-PROG-01.06 Operating Experience Program.

BP-PROG-08.01 Emergency Measures Program is also supported by a number of other Bruce Power Programs which are shown in Figure 33.

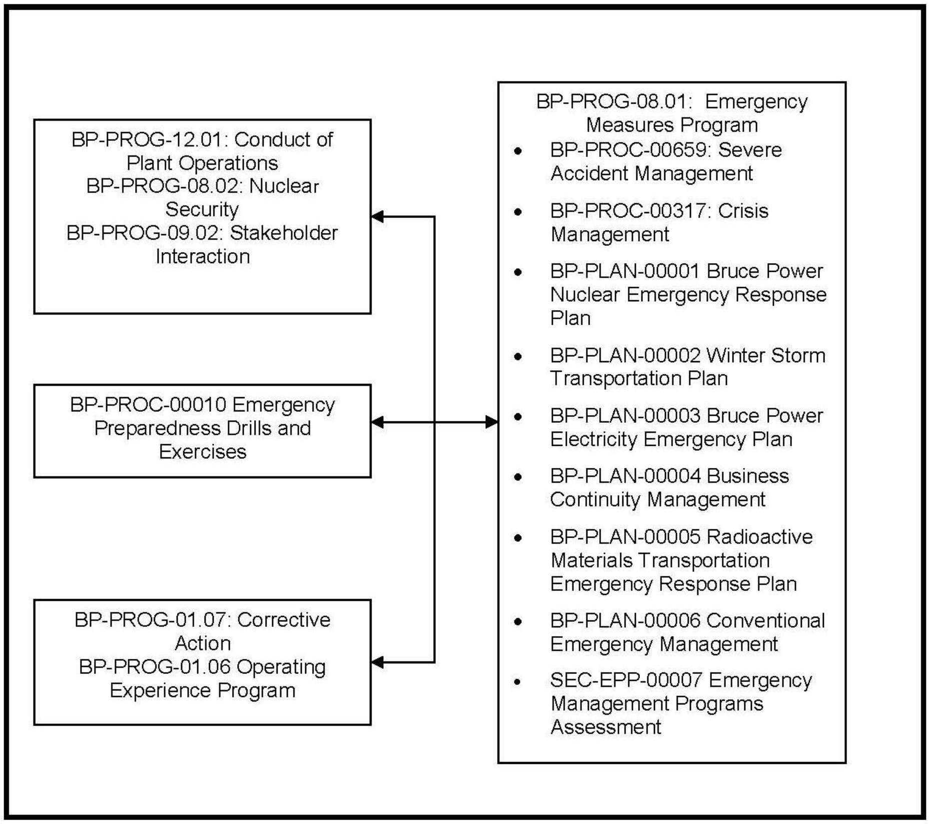

Figure 33: Overview of Governance for Emergency Planning

6.4.4.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 13 [17] [20].

General relationship of the six groups of Safety Factors have been discussed in Section 6 and illustrated in Figure 7. In this section the interfaces of SF-13 with those under 'Management' are discussed.

Organization and Administration includes all of the policies and programs in BP-MSM-1 Bruce Power Management System Manual for safe and reliable operation of Bruce A and Bruce B in accordance with the PROL. Specifically, those programs located in the left column of Figure 27 pertain to the overall organization and management of the business in support of plant operation. In this context, SF-13 Emergency Planning is a specific element of SF-10 Organization and Administration, i.e., BP-MSM-1 Bruce Power Management System Manual. As such, its implementation informs the continuous improvement of organization and administration as illustrated in Figure 34. In addition, implementation of programs and procedures identified under the SF-13 inform and support programs implemented under SF-10 Organization and Administration and SF-13 Emergency Preparedness as illustrated in Figure 34.

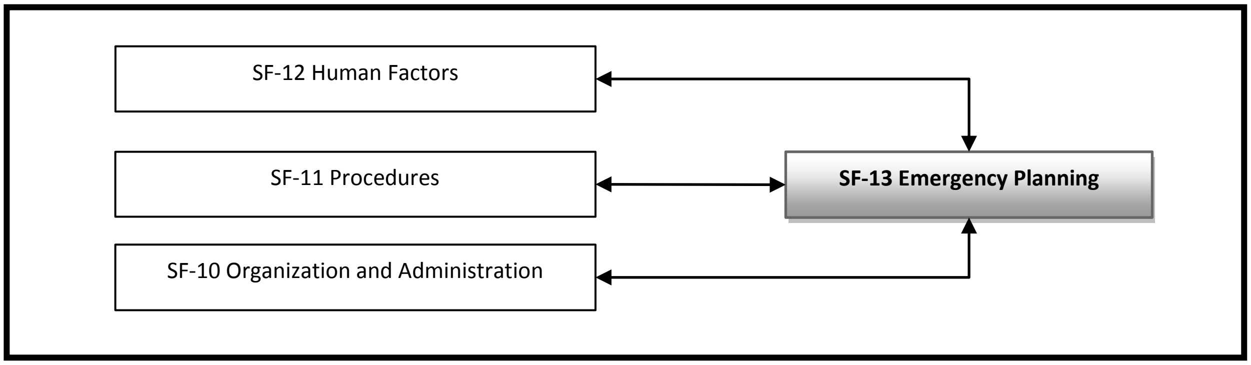

Figure 34: Safety Factor 13 Interfaces

6.4.4.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.4.4.2 are included in Section 5 of SFR 13 [17] [20].

No specific strengths were observed that are related to Emergency Planning for Bruce A during this review. A particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events for Bruce B. This strength is considered to be equally applicable to Bruce A.

The key issues (or, macro-gaps) arising from SFR 13 are provided verbatim in Table 22 and Table 23. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement are included in the IIP.

In addition, the following acceptable deviations were identified:

- CNSC REGDOC 2.10.1 (Clause 2.2.2) – Bruce A and B
- CNSC REGDOC 2.3.2 (Clause 3.3) – Bruce B
- CNSC REGDOC 2.3.2 (Clause 3.4) – Bruce B
- CNSC REGDOC 2.3.2 (Clause 4.1) – Bruce A and B
- CNSC REGDOC 2.3.2 (Clause 7) – Bruce A and B

These reviews concluded that Bruce Power meets the requirements of the Safety Factor related to Emergency Planning with the exceptions noted in Table 22 and Table 23. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The overall review indicates that the current and planned implementation of the programs related to Emergency Planning is adequate to support continued safe and reliable operation of Bruce A and B.

Table 22: Key Issues Identified for SFR 13 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF13-1	Improvements/revisions to the Emergency Measures Program, the BPNERP, and implementing documents are required, specifically: • ensuring audit findings and CNSC Action Notices are effectively addressed; • ERO Drill participation rate; • implementation of real-time off-site fixed radiological detection and monitoring; • ensuring security arrangements at off-site centres; • providing recommendations to off-site authorities; • Pre-distribution of Iodine Thyroid Blocking agents requires to be implemented (committed to CNSC by year end 2015).	Sections 5.1, 5.3.4, 7.1, 7.2.1, 7.3, 7.4 Micro-gaps against requirement clauses: REGDOC-2.10.1 – Clause 2.2.3 REGDOC-2.10.1 – Clause 2.2.4 REGDOC-2.10.1 – Clause 2.2.6 REGDOC-2.10.1 – Clause 2.3.4
SF13-2	Completion and/or resolution of Fukushima Action Items, which includes: • completion of SAMG updates to provide guidance for multi-unit severe accidents; • completion of required studies (e.g., instrumentation and equipment survivability, in-vessel retention, shield tank overpressure protection, plant habitability) in support of the first item in this list; • direct measurement combustible gas concentration or acceptable resolution of issue. (Note: resolution of FAIs is progressing according to a schedule acceptable to the CNSC).	Section 5.3.5 Micro-gaps against requirement clauses: REGDOC-2.10.1 – Clause 2.1 REGDOC-2.3.2 – Clause 3.3 REGDOC-2.3.2 – Clause 3.4 REGDOC-2.3.2 – Clause 3.5

Issue Number	Macro-Gap Description	Source(s)
SF13-3	Addressing the increased expectations for an integrated accident management program to comply with the expectation in CNSC REGDOC-2.3.2. This includes such issues as: • targeted stress tests; • effectiveness of the most suitable or preferable measures for each reactor damage state assessed and documentation in detail; • use of PRA to verify SAMG effectiveness, specification of time periods, and scenarios for training and drills; • control of contaminated run-off water to the environment.	Section 5.3.5 Micro-gaps against requirement clauses: REGDOC-2.3.2 – Clause 4.2 REGDOC-2.3.2 – Clause 4.3
SF13-4	Addressing the additional requirements in CSA N1600. There are a number of detailed additional requirements in CSA N1600 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: • an evaluation of losing critical functions, which might impact the ability to respond and recover from an emergency; • processes for deviating from emergency response plans or recovery plans; • detailed requirements for nuclear emergency recovery plans. Given that CSA N1600 is likely to be substantially revised in the short term, a phased approach should be considered for its detailed review for elements that need to be addressed by Bruce Power.	Section 5.3.5 Micro-gaps against requirement clauses: CSA N1600 – Clause 4.2.3 CSA N1600 – Clause 4.5.2 CSA N1600 – Clause 4.5.12 CSA N1600 – Clause 4.6.1 CSA N1600 – Clause 5.4

Table 23: Key Issues Identified for SFR 13 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF13-1	Addressing existing expectations for the Emergency Management Program, the BPNERP, and/or implementing documents, specifically: • ERO Drill participation rate and staff selection; • MART response timing; • Completion of the On-Site/Off-Site Emergency Response Communications Project to ensure that two independent	Sections 5.1, 7.1, 7.2.1, 7.2.1.1 7.3.2.7, 7.4 Micro-gaps against requirement clauses: REGDOC-2.10.1 – Clause 2.1 REGDOC-2.10.1 – Clause 2.2.6 REGDOC-2.10.1 – Clause 2.2.8

Issue Number	Macro-Gap Description	Source(s)
	means of communication are available to all emergency centres; • ensuring security arrangements at off-site centres; • enhancements to recovery plan framework; and • Basis for minimum shift complement and ability to respond to multi-unit events.	
SF13-2	Addressing the additional requirements in CSA N1600 and IAEA GSR Part 7. There are a number of detailed additional requirements in CSA N1600 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: • an evaluation of losing critical functions, which might impact the ability to respond and recover from an emergency; • processes for deviating from emergency response plans or recovery plans; • detailed requirements for nuclear emergency recovery plans. There are also a number of additional requirements in IAEA GSR Part 7. The more significant of these include: • for emergency workers, increased fitness for duty expectations, training, medical follow-up and psychological counselling, optimized protection • process for authorizing exceeding dose limits and obtaining qualified medical advice prior to incurring additional occupational exposure • having sufficient qualified staff manage an emergency response at all facilities if each of the facilities is under emergency conditions simultaneously	Section 5.3.5 Micro-gaps against requirement clauses: CSA N1600 – Clause 4.2.3 CSA N1600 – Clause 4.5.2 CSA N1600 – Clause 4.5.12 CSA N1600 – Clause 4.6.1 CSA N1600 – Clause 5.4 IAEA GSR Part 7 – Clause 5.49 IAEA GSR Part 7 – Clause 5.52 IAEA GSR Part 7 – Clause 5.53 IAEA GSR Part 7 – Clause 5.57 IAEA GSR Part 7 – Clause 5.60 IAEA GSR Part 7 – Clause 6.11
SF13-3	Addressing issues raised by the 2015 OSART Review: • increasing the robustness of radiation protection for on-site personnel • improving procedural guidance emergency classification • improving the radiation protection for EMC staff	Sections 5.3.1, 5.3.2, 5.3.3, 5.3.5, 7.2.2.2 Micro-gaps as identified in the Bruce B OSART Report section 9

6.5. Environment

This section summarizes the results of Safety Factor associated with the environment:

- SF-14 Radiological Impact on the Environment

6.5.1. Radiological Impact on the Environment

6.5.1.1. Objective

The objective of the review of this Safety Factor is to determine whether the operating organization has an adequate program for surveillance of the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable (ALARA).

6.5.1.2. Scope of the Review

The review has been conducted in accordance with the Bruce A ISR Basis Document [1] and Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

Verification whether the monitoring program (that provides data on the radiological impact of the nuclear power plant on its surroundings) is appropriate and sufficiently comprehensive. In particular, the review should verify that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation. (In this review task, "monitoring program" refers to both the effluent monitoring program and the environmental monitoring program.)

Additionally, as part of this review it should be verified that:

1. Concentrations of radionuclides in air, water (including river water, sea water and groundwater), soil, agricultural and marine products and animals are being monitored by the operating organization or by an independent public organization and are trended, and appropriate corrective actions are taken in the event that action levels are exceeded;
2. Potential new sources of radiological impact have been recognized by the operating organization;
3. Sampling and measurement methods are consistent with current standards;
4. Records of discharges of effluents are being monitored and trended and appropriate actions are taken to remain within established limits and to keep such discharges as low as reasonably achievable;
5. On-site monitoring is undertaken at locations and using methods that have a high probability of the prompt detection of a release of radioactive material to the environment;
6. Off-site monitoring for contamination levels and radiation levels is adequate and corrective actions are taken to keep such levels as low as reasonably achievable;

7. Actions have been taken to clean up contamination where reasonable and practicable;
8. Alarm systems to respond to unplanned releases of radioactive material from on-site facilities are suitably designed and available and will remain available in the future;
9. Appropriate data have been published on the environmental impact of the plant; and
10. Changes in the use of areas around the site have been taken into account in the development of monitoring programs.

6.5.1.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 14 [17] [19] [20].

6.5.1.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to the Radiological Impact on Environment related processes. Bruce Power's program document BP-PROG-00.02 Environmental Safety Management, defines the overall scope, business need, functional requirements, constituent elements and key responsibilities associated with the management of environmental safety. In this context, scope of BP-PROG-00.02 is not limited to radiological impacts only but covers all aspects of environmental safety. Figure 35 illustrates the relationships amongst the implementing procedures of BP-PROG-00.02 Environmental Safety Management. Other corporate level programs such as BP-PROG-01.07 Corrective Action BP-PROG-01.06 Operating Experience Program which support continuous improvement and safety culture in enhancing environmental safety performance are not shown for simplicity.

The objective of the Environmental Safety Management Program is to define the requirements and elements of environmental protection and to oversee the planning, implementation and control of activities associated with minimizing the potential adverse impact of Bruce Power operations on the natural environment by implementing elements of healthy Nuclear Safety Culture. Bruce Power's safety culture incorporates the framework of nuclear safety, industrial safety, radiological safety and environmental safety. The overall Bruce Power Environmental Safety Management Program conforms to the CNSC regulatory standards S-296[9], N286-05, as well as the International Organization for Standardization (ISO) 14001 standard for environmental management systems (EMS). Programs, processes, and procedures will, at a minimum, assure compliance with regulatory and statutory requirements and facilitate continual improvement in environmental performance.

[9] S-296 has been superseded by REGDOC 2.9.1. The Bruce Power documentation will be revised as part of the 3-yr review, per DCR 28460258.

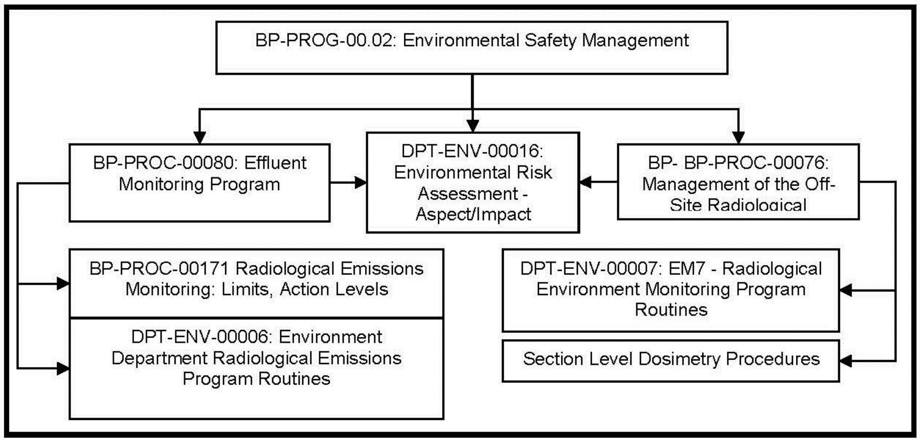

Figure 35: Overview of Governance for the Environment

6.5.1.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 14 [17] [19] [20].

General relationship of the six groups of Safety Factors including Environment have been discussed in Section 6 and illustrated in Figure 7. There will be no further discussion in this section since there is only one Safety Factor under Environment.

6.5.1.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.5.1.2 are included in Section 5 of SFR 14 [17] [19] [20].

No specific strengths were observed that are related to Radiological Impact on the Environment during this review.

The key issue (or, macro-gap) arising from Safety Factor 14 is provided verbatim in Table 24. This macro-gap a consolidation of similarly-themed micro-gaps (i.e., the "Source(s)" column) at the Safety Factor level. While the macro-gap does not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement are included in the IIP. This issue is also applicable to Bruce B.

These reviews concluded that with the exceptions noted in Table 24, Bruce Power meets the requirements of the Safety Factor related to Radiological Impact on the Environment. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The overall review indicates that the current implementation of the programs related to Radiological Impact on the Environment is adequate to support continued safe and reliable operation of Bruce A and B.

Table 24: Key Issue Identified for SFR 14 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF14-1	Performance testing of air-cleaning systems: documentation does not fully meet the requirements of CSA N288.3.4.	Section 5.8 Micro-gaps against requirement clauses: CSA N288.3.4 Clauses 8.9, 10, 11, 12, 13

6.6. Radiation Protection

This section summarizes the results of Safety Factor associated with radiation protection:

- SF-15 Radiation Protection

6.6.1. Radiation Protection

6.6.1.1. Objective

The objective of the review of this Safety Factor is defined in Appendix A of CNSC REGDOC-2.3.3 [13] and the Bruce B PSR Basis document [2] as follows:

- the extent to which radiation protection (RP) has been accounted for in the design and operation of the reactor facility
- whether RP provisions (including design and equipment) provide adequate protection of persons from the harmful effects of radiation, and ensure that contamination and radiation exposures and doses to persons are monitored and controlled, and maintained as low as reasonably achievable (ALARA).

6.6.1.2. Scope of the Review

The review is conducted in accordance with the Bruce B PSR Basis Document [2], which states that the review tasks are as follows:

- Reactor design features for RP;
- RP equipment and instrumentation for radiation monitoring;

- RP aspects during nuclear emergencies; and
- RP operating experience.

Three additional review tasks were added to align with the contents of World Association of Nuclear Operators (WANO) Guidelines for Radiological Protection at Nuclear Power Plants, WANO GL-2004 (Rev-1) for completeness and to increase the utility of the review:

- RP organization and administration;
- RP training; and
- RP Program documentation.

6.6.1.3. Regulatory Documents, Codes and Standards Assessed

The regulatory documents, codes and standards assessed in the review of this Safety Factor, along with the type of assessment conducted for each, are identified in Appendix B. The list of codes and standards is based on Section 3 of SFR 15 [20] [22].

6.6.1.4. Overview of Applicable Bruce A and B Station Programs and Processes

Bruce A and Bruce B share the same programs and procedures as applicable to Radiation Protection related processes. Radiation protection (safety) is one of the four pillars of nuclear safety which supports a healthy nuclear safety culture. BP-PROG-12.05 Radiation Protection Program defines the fundamental business needs, constituent elements, functional requirements, implementing approaches and key responsibilities associated with implementing the Bruce Power Radiation Protection Management Policy as defined in Appendix A of BP-MSM-1, Management System Manual. This Program is designed to embrace and contribute to the principles of nuclear safety as defined in BP-MSM-1, and recognizes that reactor safety, industrial safety, and environmental safety are essential to the long-term success of this Program.

BP-PROG-12.05 Radiation Protection Program sets the following objectives to meet the intent of the Bruce Power Radiation Protection Management Policy:

1. Ensure public and occupational exposures to ionizing radiation are controlled such that:
    a. Individual doses are kept below regulatory dose limits.
    b. Unplanned exposures are avoided.
    c. Individual and collective doses are maintained at levels As Low as Reasonably Achievable (ALARA), social and economic factors being taken into account.
2. Control the movement of people and materials in a manner that prevents the uncontrolled release of contamination or radioactive materials from Bruce Power facilities.

3. The achievement of high standards of radiation protection performance in accordance with industry best practices and the WANO Guidelines for Radiological Protection at Nuclear Power Plants, WANO GL 2004-01 (Rev-1).
4. Ensure compliance with Canadian Nuclear Safety Commission (CNSC) Regulations, Licences and Canadian Standard Association (CSA) requirements pertaining to contamination control and radiation protection, specifically CSA N286-05, Management System Requirements for Nuclear Power Plants, Section 6.24.

This Program also defines the requirements for compliance with Ontario Occupational Health and Safety Act (OHSA), X-Ray Safety and Radiation Emitting Devices (RED) Act requirements. These regulations pertain to x-ray generating equipment not licensed by the CNSC.

Elements of Radiation Protection Program and Associated Level 2 Radiation Protection Procedures are listed in Section 4 of SFR 15 [20] [22].

Figure 36 illustrates the relationships amongst the Level 2 implementing procedures of BP-PROG-12.05 Radiation Protection Program. Other corporate level programs, such as BP-PROG-02.01 Worker Staffing, BP-PROG-01.07 Corrective Action, BP-PROG-01.06 Operating Experience Program, which support continuous improvement and safety culture, are not shown for simplicity.

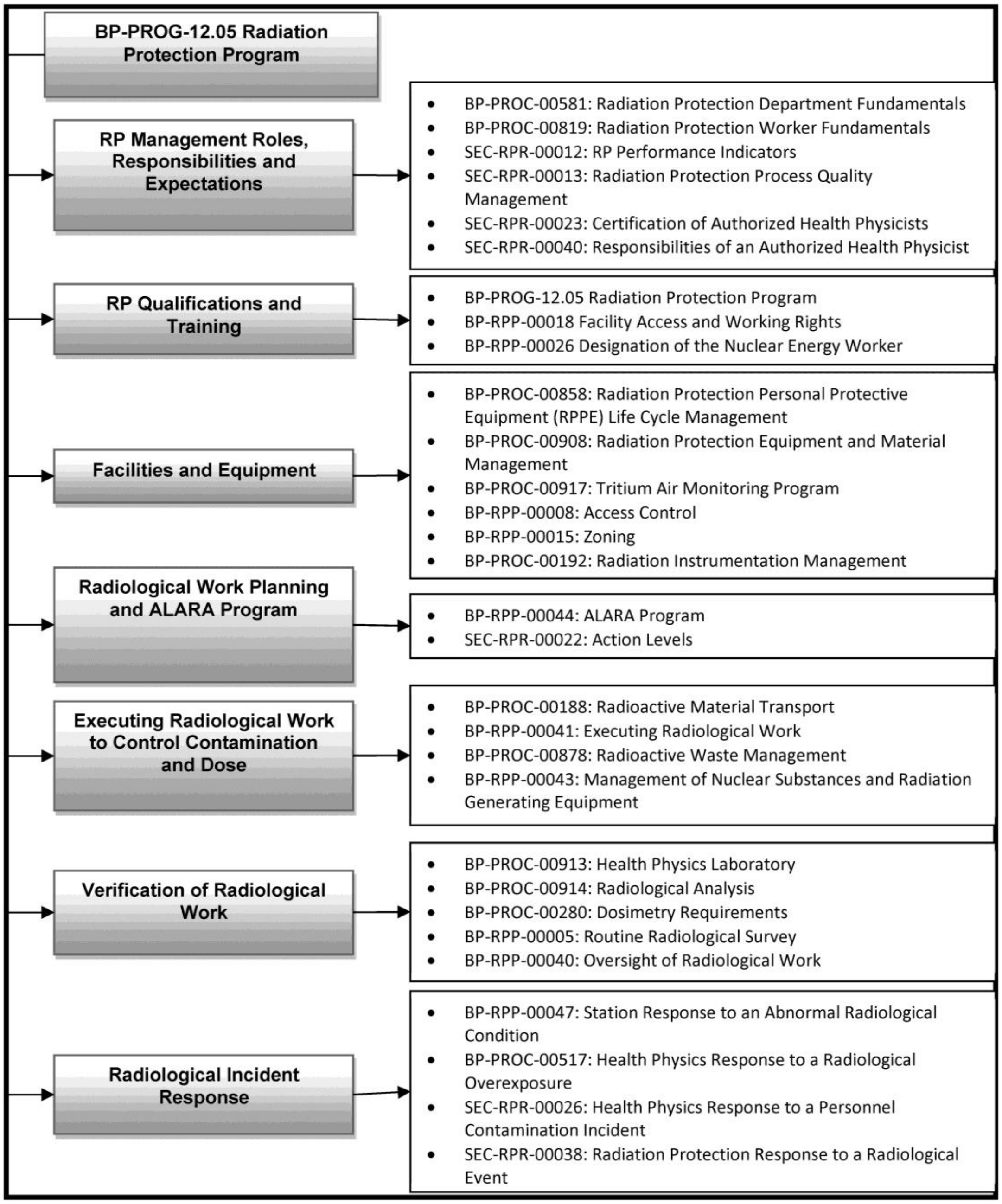

Figure 36: Overview of Governance for Radiation Protection

6.6.1.5. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce PSR. Those aspects that are addressed in this Safety Factor, or where more detail is provided in other Safety Factor Report(s) are summarized in Section 6 of SFR 15 [20] [22].

General relationship of the six groups of Safety Factors including Radiation Protection have been discussed in Section 6 of SFR 15 and illustrated in Figure 7. There will be no further discussion in this section since there is only one Safety Factor under Radiation Protection.

6.6.1.6. Summary and Conclusions

Results of the assessments for each review task listed in Section 6.6.1.2 are included in Section 5 of SFR 15 [20] [22].

Bruce Power has a mature and comprehensive radiation protection program that, by 2009, had begun to show the effects of aging and lack of maintenance. This contributed to the loss of radiation protection controls observed during the 2009 Alpha Contamination Incident. Since that time, Bruce Power has made progress in addressing the deficiencies through RP improvement and excellence programs. Bruce Power recognized that significant change was required in all areas of RP at Bruce Power, and acted on this by developing extensive RP improvement initiatives and significantly reorganizing the RP Department at each of the Bruce Power facilities.

Bruce Power has since achieved top ranked status for Collective Radiation Exposure (CRE) in North America. This industry-leading CRE performance has been identified as a strength in performance.

The key issues (or, macro-gaps) arising from SFR 15 are provided verbatim in Table 25 and Table 26. These macro-gaps are consolidations of similarly-themed micro-gaps (i.e., the “Source(s)” column) at the Safety Factor level. While the macro-gaps do not progress beyond the Safety Factor Reports, the micro-gaps are evaluated as part of the Global Assessment and as appropriate practicable opportunities for improvement are included in the IIP.

In addition, the following acceptable deviations were identified:

- WANO GL 2004-01 (Clause I.C10) – Bruce A and B
- WANO GL 2004-01 (Clause III.C3) – Bruce A and B
- WANO GL 2004-01 (Clause IV.C1) – Bruce A and B.

These reviews concluded that with the exception of micro-gaps noted in Table 25 (which supersede those shown in Table 26), Bruce Power meets the requirements of the Safety Factor related to Radiation Protection. Practicable improvements to resolve the identified micro-gaps will enhance compliance to a level similar to those required for modern plants. The overall review indicates that the current implementation of the programs related to Radiation Protection is adequate to support continued safe and reliable operation of Bruce A and B.

Table 25: Key Issues Identified for SFR 15 – Bruce A

Issue Number	Macro-Gap Description	Source(s)
SF15-1	ALARA Program The ALARA Program documentation is inconsistent with WANO guidance in the areas of: documentation of ALARA Committee TOR; ALARA incentive program; and Radiation Exposure Permits. There is misalignment between ALARA planning and outage planning target dates.	Section 5.1.1.3 Micro-gaps against WANO GL 2004-01 guidance clauses: I.C5 (Gap 1) V.C1 (Gap 1) V.C2 (Gap 1) VII.C2 (Gap 1, Gap 2)
SF15-2	Radiological hazard control There is one noted discrepancy against the guidance regarding response to airborne radiological hazard control.	Section 5.4.1 Micro-gaps against WANO GL 2004-01 guidance clause: IV.C2 (Gap 1)
SF15-3	RP equipment and instrumentation There is no documented lifecycle management process for the FAGM system. The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation.	Section 5.2.2 Programmatic Gap (Gap1) Effectiveness Gap (Gap 1) Micro-gaps against WANO GL 2004-01 guidance clauses: VI.C2 (Gap1, Gap 2, Gap 3, Gap 4, Gap 5, Gap 6)
SF15-4	Organization and administration There are instances when use of the action tracking process did not result in resolution of the identified issues.	Section 5.4.1 Effectiveness Gap (Gap 1)
SF15-5	RP Program Documentation There are instances of unclear standards in the RP Program, and current RP practices are not always documented in RP Program governance: RP Programs Manager role; dose reporting requirements; dismantling objects to survey inaccessible surfaces for contamination; confirmation that there is no unexpected dose received outside the Controlled Area; back-out criteria for DRPs and airborne particulates; gamma-sensitive whole-body monitors at RCA exits; training on the use of CATS; secure covering of HEPA units in storage.	Sections 5.1.1.3, 5.1.3.3 and 5.6.1 Programmatic Gap (Gap1, Gap 2) Micro-gaps against WANO GL 2004-01 guidance clauses: I.C2 (Gap 1) I.C4 (Gap 1) VI.C3 (Gap 1)

Table 26: Key Issues Identified for SFR 15 – Bruce B

Issue Number	Macro-Gap Description	Source(s)
SF15-1	ALARA Program The ALARA Program documentation is inconsistent and lacking some recommendations made in the guidance in the areas of: conduct of ALARA Committees; ALARA incentive program; and Radiation Exposure Permits. There is misalignment between ALARA planning and outage planning target dates.	Section 5.1.1 Micro-gaps against WANO GL 2004-01 guidance clauses: I.C5 (Gap 1) V.C1 (Gap 1) V.C2 (Gap 1) VII.C2 (Gap 1, Gap 2)
SF15-2	Radiological hazard control There are noted discrepancies against the guidance in the following areas of the radiological hazard control program: LHRA controls; airborne radioactivity; and restriction of contamination prone materials in the controlled area.	Sections 5.1.2 and 5.1.3 Micro-gaps against WANO GL 2004-01 guidance clauses: III.C1 (Gap 1) IV.C2 (Gap 1) VI.C2 (Gap 1) VI.C3 (Gap 1, Gap 2)
SF15-3	RP equipment and Instrumentation There are noted gaps in the adequacy and condition of RP equipment and instrumentation when compared against the WANO recommendations and REGDOC-2.3.3 RP review tasks.	Section 5.2.2 Programmatic Gaps (Gap1, Gap 2) Effectiveness Gap (Gap 1) Micro-gaps against WANO GL 2004-01 guidance clauses: VI.C2 (Gap 2, Gap 3, Gap 4, Gap 5)
SF15-4	Organization and administration There are instances of ineffective use of the action tracking process to address RP issues.	Section 5.4.1 Effectiveness Gap (Gap 1)
SF15-5	RP Program Documentation There are instances of ineffective management of RP Program standards, and current RP practices are not always documented in RP Program governance: RP Manager roles; reporting lower-level significance OPEX externally; dose reporting requirements; dismantling objects to survey inaccessible surfaces for contamination; confirmation that there is no unexpected dose received outside the Controlled Area; back-out criteria for DRPs and airborne particulates; gamma sensitive whole body monitors at RCA exits; training on the use of CATS.	Section 5.6.1 Micro-gaps against WANO GL 2004-01 guidance clauses: I.C2 (Gap 1) I.C4 (Gap 1)

Part III: Global Assessment

Section	Title
7	Consolidation of Safety Factor Findings
8	Classification of Safety Factor Findings and Development of GIs
9	Improvement Initiatives Outside PSR
10	Definition of Global Improvement Opportunities
11	Prioritization and Ranking of GIOs

Appendix	Title
Appendix D	Category 1: No Reasonable and Practicable Improvements can be made
Appendix E	Category 2: Safety Improvement Considered Unnecessary to Implement as Part of IIP
Appendix F	Category 3: Safety Improvement In-Progress
Appendix G	Category 4: Safety Improvement Considered Necessary

7. Consolidation of Safety Factor Findings

The objective of consolidation of Safety Factor review findings is to:

- Address any overlaps, omissions, and interface issues of the findings from the SFRs; and
- Link all related micro-gaps where appropriate.

The findings from each Safety Factor review, whether strengths or micro-gaps (negative findings) are based on the fairly narrow perspective of the Safety Factor. This step of global assessment provides for the consolidation of these findings to establish global findings through the removal of duplication and the broadening of context to make the findings comprehensive. This applies both to the strengths, as well as the individual micro-gaps or macro-gaps (collection of negative findings). The consolidation of Safety Factor findings is described in Section 7.1.

Micro-gaps across all SFRs, summarized in Section 6, were already uploaded in the PSR database. There were a total of four-hundred and forty-one (441) micro-gaps identified across all SFRs. Each micro-gap was provided with database identification. The following information was included for each micro-gap:

- SFR Number
- Macro-gap Number and Title (as applicable)
- Reference regulatory document, code or standard
  - Applicable section or clause
  - Text of the requirement relevant to the micro-gap
- Description of the micro-gap
- Type of the micro-gap (requirement, guidance, etc.)
- Mapping to the CNSC Safety and Control Area (or another appropriate place)

7.1. SFR Micro-Gap Consolidation

The purpose of this step is to consolidate individual micro-gaps across all SFR findings by identifying common micro-gaps, thereby eliminating duplication and identifying potential omissions of micro-gaps. In terms of consolidation:

- Duplication occurs as a result of micro-gaps that have been identified in different SFRs which are same or similar. The major reason for duplication is assessment of same or similar requirements or review tasks across different Safety Factors and using the same PSR process.
- Potential omissions of micro-gaps may occur as a result of differences in sets of regulatory documents, codes and standards and review tasks defined in the PSR basis

documents, as well as the available information at the freeze dates identified in the ISR and PSR Basis Documents.

As described in Section 5.2.1 the approach used was to look at two aspects first from a requirement perspective then secondly from a process perspective.

The remaining micro-gaps were scrutinized for coverage of the same topic or process. Although a number of micro-gaps were observed topically similar, they were not duplicates.

One-hundred and thirty (130) sets of micro-gaps were identified as the same or similar during the consolidation of micro-gaps.

- 24 sets of micro-gaps out of 74 micro-gaps were identified as the same or similar, and 11 identified as unique in Appendix D;
- 35 sets of micro-gaps out of 113 micro-gaps were identified as the same or similar, and 33 identified as unique in Appendix E;
- 54 sets of micro-gaps out of 178 micro-gaps were identified as the same or similar, and 44 identified as unique in Appendix F;
- 17 sets of micro-gaps out of 76 micro-gaps were identified as the same or similar, and 37 identified as unique in Appendix G.

This review also helped in forming a list of micro-gaps that are topically similar which was used in development of GIs in Section 10.

8. Classification of Safety Factor Findings and Development of Global Issues

8.1. Introduction

The purpose of this step is to assess consolidated micro-gaps from SFRs and group them into Global Issues based on their topical similarities.

Micro-gaps are assessed for their safety significance and priority as part of the Global Assessment (GA) and for consideration in the IIP.

Input for this step is the set of consolidated micro-gaps in Section 7.1. The assessment was based on the guidance provided in Section 5.3.

8.2. Assessment and Classification Scheme and Results

Each micro-gap was assessed and then classified under one of the four groups described in Section 5.3.

- Category 1- No reasonable and practicable improvements can be identified
- Category 2- Safety improvements considered unnecessary to implement as part of IIP
- Category 3- Safety improvements in-progress
- Category 4- Safety improvements considered necessary

The results of classification for each category are documented in the following sub-sections.

8.2.1. Category 1: No reasonable or Practicable Improvements can be Identified

Negative findings in this category generally result from comparison against modern codes and standards and some international practices that have not been incorporated in the PROL. Some examples are:

- A generic requirement which results in fundamental design changes to SSCs of the plant as a whole which cannot be accommodated within the current configuration of SSCs and plant layout. Due to the existing coupling of SSCs and their functional capabilities in the current design of the plant, changes to SSC(s) would also impact other physically connected or functionally related SSCs. Normally, compliance with this type of new requirement or principle can only be practically dealt with for a new plant, as the physical and functional relationships have to be defined first to meet the high level regulatory dose limits, safety goals, associated classification of SSCs and consequently as design requirements. For example, the following principles and requirements would be considered in this category:

- More conservative dose limits or safety goals, or higher safety margins or combination thereof than those currently in place due to new or updated requirements;
- New design requirements which were not considered in the original design of the plant, e.g., physical changes driven by evolving design philosophy for new NPPs as additional requirements or newer interpretation of principles, such as redundancy, diversity, separation in terms of DID or improvement of safety goals;
- Changes to the classification of SSCs or events or event sequences which lead to different or new design requirements in the current design basis of SSCs

- A practice that is not adopted by either the CNSC or the Industry in Canada for operating plants; e.g., requirements applicable to new NPP or a different design technology, such as an LWR.
- A requirement that is not adopted by neither the CNSC nor the Industry in Canada that fundamentally impacts the organization of the plant, its governance and processes which is not sustainable in terms of business objectives

For micro-gaps in this category reason(s) for the classification have been documented and the issue will be revisited after an appropriate period of time (for example at the next PSR).

Integrated impact of not implementing these micro-gaps is assessed in Part III: Global Assessment.

There were a total of 74 micro-gaps out of 441 classified as ‘impracticable’. Results of the reviews together with reason(s) for the classification are presented in Appendix D.

As discussed in Appendix D, in many cases, the assessment of the current design demonstrated that there are other provisions in the design and operation that address the new requirement(s). Given the above considerations, and that both Bruce A and Bruce B Safety Reports demonstrate that the current licensing limits are met with adequate safety margins, individual micro-gaps which are classified in this category were judged to have a low safety significance, and would have high resource usage to realize the marginal benefits. These micro-gaps could not result in meeting licensing limits applicable to new plants and deliver the expected marginal safety benefit but only when implemented collectively. However, implementing these micro-gaps collectively would be akin to building a new plant which is beyond the objectives and scope of the PSR process.

In addition, when taken as a whole, the number of impracticable micro-gaps that require fundamental design changes to specific SSCs and the plant as a whole, and their integrated impact with potentially conflicting physical and layout constraints, cannot be accommodated within the current configuration of SSCs and plant layout. Due to the existing coupling of SSCs and their functional capabilities in the current design of the plant, changes to SSC(s) would also impact other physically connected or functionally related SSCs. This integrated impact will further increase the level of complexity and impracticability.

8.2.2. Category 2: Safety Improvement Considered Unnecessary to Implement as Part of IIP

Micro-gaps in this category are from three sources:

- Micro-gaps resulting from comparison against modern codes and standards and some international practices where there are alternative ways of addressing them within the current licensing framework and industry best practices. For micro-gaps in this category, reason(s) for the classification are documented as appropriate, and any follow-up actions or oversight is documented including associated ARs (Action Requests) for its implementation and the issue is categorized as “Closed” in the PSR database.
- Micro-gaps where safety improvements afforded by addressing them would be rendered unessential because the current DID provisions and level of safety are sufficiently robust and its contribution to dose acceptance criteria and overall safety goals may be insignificant. For micro-gaps in this category, reason(s) for the classification are documented and the issue is categorized as “Closed” in the PSR database.
- Individual micro-gaps resulting from less than adequate implementation of the current governance and associated procedures. These are mostly identified during the review of audits, FASAs, peer reviews as part of review task assessments and in most cases specific corrective actions have already been identified for addressing them are in progress. In this context, they do not present a generic process improvement opportunity that is safety significant and can be dealt with through the current Corrective Action processes in place as appropriate.

There were a total of 113 micro-gaps out of 441 classified as ‘Not Necessary’ to implement as part of the IIP. Results of the reviews together with reason(s) for the classification are presented in Appendix E.

In addition, the CNSC identified 18 specific items as gaps based on the CNSC reviews of the SFRs and the GAR/IIP (Revision R01). The SFRs are not being re-issued, and therefore these additional gaps will not be incorporated in the PSR database. Rather, they will be addressed in a manner analogous to the SFR-generated Category 2 gaps, in that Bruce Power will establish an AR number that will be used to track how each is being addressed, and will provide the AR numbers to the CNSC. These additional gaps are captured in a separate table in Appendix E.

8.2.3. Category 3: Safety Improvement in Progress

Micro-gaps that are the same as those that have already been identified in the previous PSRs or by other means are included in this category if there are initiatives or commitments in place to resolve them. Each micro-gap was checked against the status of the current IIP [23] initiatives or commitments, such as CNSC regulatory commitments, management actions, capital projects in place. Status of the applicable corrective actions in place was investigated and documented.

Each review resulted in one of the three sub-categories for such micro-gaps:

- If the associated corrective action(s) is completed, appropriate references pertaining to the completion is provided and the issue was considered as “Closed”. There were 3 micro-gaps in this sub-category.
- If the associated corrective action(s) is in progress and being reported to the CNSC as part of the current IIP, appropriate references pertaining to the status was provided. Hence, such micro-gaps are not considered further during the GA, and will be retained in the IIP in their current form. There were a total of 175 micro-gaps in this category.
- If the associated corrective action(s) is in progress but not being reported the CNSC as part of the current IIP [11], appropriate references pertaining to the status is provided and the issue is considered as “In-Progress”. Such micro-gaps are included in Section 9 and included in the IIP directly. There were no micro-gaps in this sub-category.

There were a total of 178 micro-gaps out of 441 classified as ‘In-Progress’. Results of the reviews together with reason(s) for the classification are presented in Appendix F.

8.2.4. Category 4: Safety Improvement Considered Necessary

This category includes the remaining micro-gaps from Categories 1, 2 and 3 described above. These micro-gaps are considered as those where safety improvements are necessary. Generally these include maintenance, repair, replacement of plant SSCs important to safety and reliability, engineering assessments and analyses supporting continued operation for the assessment period, practicable design modifications and improvements to ensure compliance with the current design basis and expectations in the modern codes and standards, as well as updating/improving or extending of plant documentation or operating procedures.

A total of 76 micro-gaps out of 441 were identified as ‘Necessary’ to be considered in the development of the IIP. The distribution of the micro-gaps that are identified as ‘Necessary’ across all SFR micro-gaps is as follows:

- SF-1: 14 micro-gaps
- SF-3: 1 micro-gaps
- SF-4: 2 micro-gaps
- SF-5: 2 micro-gaps
- SF-8: 2 micro-gaps
- SF-12: 9 micro-gaps
- SF-13: 31 micro-gaps
- SF-14: 6 micro-gaps
- SF-15: 9 micro-gaps

Results are presented in Appendix G.

These micro-gaps were used as input for the development of Global Issues in Section 8.2.5.

8.2.5. Development and Ranking of Global Issues

The purpose of this step is to develop and rank Global Issues (GIs) making use of the insights gained in Section 7.1 where micro-gaps that are same were identified and topical similarities were observed. Individual micro-gaps classified as Category 4 in Section 8.2.4 are reviewed against each other for common features thereby grouping them as sets of Global Issues. The approach used was as described in Section 5.3 reviewing two aspects of each micro-gap; first from common or similar requirement(s) perspective and if no micro-gap(s) are identified then secondly from a common process perspective.

At the end of this step all 76 micro-gaps identified in Section 8.2.4 were mapped in one of the GIOs developed for Bruce A and Bruce B. Each GI was ranked at Tier 2 of the Value Tree as described in Section 5.6 and Appendix C.

Development of GIs was performed in the PSR database. A specific verification was performed to ensure that all Category 4 micro-gaps are linked to a GI.

The Global Issues and their associated CARDs and micro-gaps are listed in Appendix G. Table 27 provides the list of GIs, as well as the CNSC Safety and Control Area to which they are associated.

Table 27: List of Global Issues

GIO No.	CNSC S&C Area	GIO TITLE
GIO-001	Physical design	Improve documented design basis
GIO-002	Physical design	Implement design changes to improve severe accident response
GIO-003	Physical design	Assess pipe whip and jet impingement
GIO-005	Physical design	Assess cyclic loads of pressure retaining components designed per ASME III or VIII
GIO-009	Safety analysis	Update safety analysis to align with REGDOC-2.4.1
GIO-011	Operating performance	Implement enhancements to SAMG
GIO-019	Physical design	Assess and improve seismic qualification
GIO-024	Management system	Enhanced Periodic Safety Review to Support Asset Management
GIO-025	Fitness for service	Perform R&D in support of fuel channel life cycle management initiatives

GIO No.	CNSC S&C Area	GIO TITLE
GIO-026	Physical design	BA & BB New Neutronic Trips
GIO-028	Fitness for service	Upgrade Emergency and Standby Power Supplies
GIO-034	Fitness for service	Safety System Reliability
GIO-036	Physical design	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
GIO-037	Physical design	Document design basis for zoning and shielding
GIO-039	Fitness for service	Equipment Reliability and Maintenance
GIO-043	Human performance management	Validation of Human Credited Actions
GIO-044	Emergency management and fire protection	Emergency preparedness
GIO-056	Fitness for service	Fuel Channel Replacement
GIO-057	Fitness for service	Steam Generator Replacement
GIO-058	Fitness for service	Feeder Replacement
GIO-059	Fitness for service	Calandria and Shield Tank Assembly Major Inspection
GIO-060	Fitness for service	Preheater Inspections
GIO-062	Fitness for service	PHT Pump Seal Bellows Replacement
GIO-064	Fitness for service	Control Distribution Frame (CDF) Terminal Replacement
GIO-065	Fitness for service	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection
GIO-066	Fitness for service	Pressurizer and Supports- Internal Inspection
GIO-070	Fitness for service	Air Operated Valves-Replacement
GIO-071	Fitness for service	Large Motors-Refurbishment/Replacement
GIO-076	Fitness for service	DCC Cables and WIBAs –Replacement
GIO-077	Fitness for service	Moderator Heat Exchangers- Replacement
GIO-078	Fitness for service	Maintenance Cooling Heat Exchanger- Replacement

GIO No.	CNSC S&C Area	GIO TITLE
GIO-081	Physical design	Human Factors in Design of Nuclear Power Plants
GIO-082	Environmental protection	Performance testing of nuclear air-cleaning systems
GIO-083	Safety analysis	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
GIO-086	Fitness for service	PHT Valves-Refurbishment of 33120-MV23
GIO-088	Management system	Improve Licencing Processes
GIO-089	Safety analysis	Whole-Site Probabilistic Risk Assessment
GIO-090	Physical design	SDS2 Enhancements
GIO-091	Physical design	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
GIO-092	Physical design	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
GIO-093	Radiation protection	RP equipment and instrumentation maintenance and life cycle management
GIO-094	Radiation protection	Effective use of the action tracking system in Radiation Protection
GIO-095	Fitness for service	45VDC Power Supplies-Replacement
GIO-097	Physical design	Bruce A Legacy Registration- Implementation Projects
GIO-098	Physical design	Bruce B Legacy Registration- Implementation Projects
GIO-099	Physical design	Install Correctly Sized Maintenance Cooling Relief Valves
GIO-100	Physical design	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	Physical design	M/34720 Relief Valves For Overpressure Protection
GIO-102	Physical design	I/63472 Remote Relief Valve Position Indication
GIO-103	Fitness for service	Implementation of Asset Management Activities
GIO-104	Fitness for service	Ongoing Work on Bruce B Heat Transport Vibration Project

9. Improvement Initiatives Outside PSR

The purpose of this step is to collect and integrate all non-SFR initiatives that have been identified through other assessments or initiatives outside the PSR to be considered in the IIP based on input from Bruce Power following the process described in Section 5.4.1.

The result of this step is a list of consolidated safety-related improvement initiatives as part of:

- IIP initiatives that are in progress (in this case the 2014 IIP [11]);
- Additional MCR and Asset Management initiatives to be included in the IIP based the screening process described in Section 5.4.1; and
- Other initiatives to be included, based on input from Bruce Power.

9.1. IIP Initiatives In-Progress

The approach used results in the integration of the latest IIP [23] submitted to the CNSC with the practicable safety improvements identified in this PSR. This process also allows for augmentation of the list of safety-related improvement initiatives to be considered for Global Assessment (GA) and IIP on a continuous basis reflecting Bruce Power's long-term plans and commitments for safe and reliable operation of Bruce A and Bruce B beyond the current PSR or PROL as illustrated in Figure 2.

The 11 GIOs included in the 2014 IIP [23] that are in progress are shown (by rank) in Table 28.

Table 28: List of GIOs In-Progress from IIP – 2014

GIO	CNSC S&C Area	GIO TITLE
GIO-028	Fitness for service	Upgrade Emergency and Standby Power Supplies
GIO-025	Fitness for service	Perform R&D in support of fuel channel life cycle management initiatives
GIO-019	Physical design	Assess and improve seismic qualification
GIO-009	Safety analysis	Update safety analysis to align with REGDOC-2.4.1
GIO-011	Operating performance	Implement enhancements to SAMG
GIO-001	Physical design	Improve documented design basis
GIO-002	Physical design	Implement design changes to improve severe accident response
GIO-026	Physical design	BA & BB New Neutronic Trips
GIO-024	Management system	Enhanced Periodic Safety Review to Support Asset Management

GIO	CNSC S&C Area	GIO TITLE
GIO-003	Physical design	Assess pipe whip and jet impingement
GIO-005	Physical design	Assess cyclic loads of pressure retaining components designed per ASME III or VIII

The following GIOs were reported as complete in the last update of the 2014 IIP [23]:

- Improve Fire Protection provisions to achieve alignment with N293-07 requirements;
- Improve Emergency Response capability; and
- Fire Training Area replacement.

9.2. MCR and Asset Management Initiatives Included in IIP

Section 5.4.1 describes the guidance on how to screen activities supporting safe long term operation from Bruce Power's MCR and Asset Management Plans for consideration in the GA and IIP.

This process was applied to the Unit 6 MCR items, based on the list of initiatives provided by Bruce Power. Table 29 and Table 30, respectively, provide those items that are included in the IIP and those that are screened out. For each item, the scope ID, description and applicable units are provided:

- Table 29 provides the 17 items that are included in the IIP.
- Table 30 provides the 52 items that have been screened out.

Table 29: MCR Scope Included in IIP

Scope ID	Description	Unit
34	DCC Cables and WIBAs -Replacement	3 to 8
44	Moderator Heat Exchangers- Replacement[10]	3 to 8
46	Maintenance Cooling Heat Exchanger- Replacement	3 to 8
170	45VDC Power Supplies-Replacement	3 to 8
285	Large Motors – Maintenance Cooling System (MCS) Pump Motors	3 to 8
345	Air Operated Valve- Nuclear Valve Replacement	3 to 8
347	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves	3 to 8

[10] The Moderator HXs for Unit 6 have been found to be in good shape. Based on that, the Moderator HXs in Units 5, 6 and 7 will be replaced, but outside of the respective MCR outages.

Scope ID	Description	Unit
4	Calandria and Shield Tank Assembly Major Inspection	3 to 8
11	Preheater Inspections	3 to 8
14	PHT Valves-Refurbishment of 33120-MV23	3 to 8
16	PHT Pump Seal Bellows Replacement	3 to 8
57	Control Distribution Frame (CDF) Terminal Replacement	3 and 4
92	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection	3 to 8
94	Pressurizer and Supports- Internal Inspection	5 to 8
38826	Replacement of Fuel Channels	3 to 8
38827	Replacement of Steam Generators	3 to 8
38828	Replacement of Feeders	3 to 8

Table 30: MCR Scope Not Included in IIP

Scope ID	Description	Unit
30	Varian V72 Digital Control Computers(DCC)- Replacement	3 to 8
32	DCC Analog Input (AI) -Replacement	3 to 8
53	Shutdown Cooling Heat Exchangers- Replacement	3 to 8
69	Medium Voltage Buried or Underground Power Cables-Refurbishment or Replacement	3 to 8
73	Medium Voltage Remainder Power Cables-Refurbishment or Replacement	3 to 8
74	Manual Valves-Refurbishment or Replacement	3 to 8
160	Primary Irradiated Fuel Bay (PIFB) Heat Exchangers- Replacement	3 to 8
161	Secondary Irradiated Fuel Bay (SIFB) Heat Exchangers- Replacement	3 to 8
168	Pressure Vessels and Tanks	3 to 8
172	Control Relays- Bruce B	3 to 8
184 (AMOT)	Isolated Phase Bus (IPB) and Cooling Components	3 to 8
200	High Voltage Power Cables-Refurbishment or Replacement	3 to 8
251	Main Boiler Feed Pump Motors-Refurbishment or Replacement	3 to 8
256	Circulating Cooling Water (CCW) Pumps -71210-P1/2/3	3 to 8

Scope ID	Description	Unit
257	Circulating Cooling Water (CCW) Pump Motors -71210-PM1/2/3	3 to 8
265	Large Motors - Low Pressure Service Water (LPSW) Pump Motors	3 to 8
268	Condenser Extraction Pumps (CEP) 43210-P1/2/3 Overhaul	3 to 8
281	Large Motors – Shut Down Cooling (SDC) Pump Motors	3 to 8
293	Large Motors – Pressurizing Pump Motors	3 to 8
322	Turbine Electro-hydraulic Governing and Turbine Supervisory Bridging Strategy	3 to 8
323	Turbine Electro- hydraulic Governing and Supervisory System Replacement	3 to 8
351	Air Operated Valve – BB ECI NV Interspace Test MVs	3 to 8
359	Air Operated Valve- Moderator Purification MV80/81- Replacement	3 to 8
360	Air Operated Valve- Liquid Zone Control Valve Replacement	3 to 8
367	Horizontal In-Core Flux Detectors- SDS2	3 to 8
371	Horizontal Ion Chamber Detectors- SDS2	3 to 8
379	Vertical In-Core Flux Detectors- SDS1 & RRS	3 to 8
383	Vertical Ion Chamber Detectors- SDS1 & RRS	3 to 8
13	PHT Pump Journal Inspections and Seal Replacements	3 to 8
50	Process Control Devices in PHT Feed, Bleed and Relief System	3 to 8
99	Liquid Zone Control (LZC) Recombination Unit (6-34810-RU1)- Catalyst Replacement	3 to 8
103	Low Pressure (LP) Turbine Inspections	3 to 8
105	Feedwater Heater Inspections	3 to 8
126	Turbine Valve Inspections/Overhaul	3 to 8
184 (MCR Scope ID)	Moderator Cover Gas (MCG) Recombiner Inspections and Catalyst Replacement	3 to 8
188	Moderator Purification Commissioning Filter-Installation	3 to 8
191	New Core Start-up Instrumentation	3 to 8
196	Transfer buses	0B
264	Low Pressure Service Water (LPSW) pumps 5/6/7/8-71310-P1/2/3/4	5 to 8
288	The Primary Heat Transport (PHT) pumps 5/6/7/8-33120-P1/2/3/4	5 to 8
305	Guelph (Kerotest/Taylor Forge) Swing Check Valves (ones not covered under USI specific templates)	5 to 8
334	600V Circuit Breakers	0B, 5 to 8

Scope ID	Description	Unit
71	Bruce B Medium Voltage High Priority Power Cables - approx. 146 Cables	0B, 5 to 8
88	Emergency Water System Piping- 71380 Above-Ground Piping and Pipe Components (Elbows, Reducers, Diffusers, etc.)	0B, 5 to 8
91	Low Pressure Service Water Piping - 71310 Above-Ground Piping and Pipe Components (Elbows, Reducers, Diffusers, etc.)	0B, 5 to 8
180	Protective Relaying (MPO, SG/EPS/EPG, CL III/IV & CL I/II)	0B, 5 to 8
192	13.8 kV Class III and Class IV Switchgear Buses	0B, 5 to 8
326	Environmentally Qualified Cables	0B, 5 to 8
215	Primary Heat Transport (PHT) Transformers	5 to 8
269	Large Motors – Condensate Extraction Pumps (CEP) Motors	5 to 8
277	Large Motors – Moderator Main Pump Motors	5 to 8
309	Stator Maintenance Program	5 to 8

9.3. Other Initiatives Included in IIP

Table 31 provides the list of other initiatives to be included in the IIP, based on input from Bruce Power. A number of CNSC commitments that are included in the IIP are those projects related to improvements in Fire Protection. These are taken from Reference [33].

Table 31: List of Other Initiatives Included in IIP

Title
Bruce B Main Control Panel PL18A Upgrade (This project has now been completed, and therefore not included in the IIP)
BA ASB Fire Protection Upgrades
BB U0 Fuel Storage Area Sprinkler Upgrades
Bruce B Fireworks Terminal Replacement
Unit 1 and 2 Fire Upgrades (Restart)
Bruce B Firewater Pipe Replacement
BA Standby Generator Building Fire Protection Upgrade
Bruce B Fire Detection Upgrade
Bruce B VESDA Upgrade

Title
Bruce B Fire Barriers (Cable Wrap) upgrades
Air Foam System Replacement
BB Standby Generator Building Fire Protection Upgrade
Bruce A Fire Barriers Upgrades
BB EPG / EWPS Building Fire Protection Upgrade
BB Maintenance Cooling Interspace Protection
Unit 1 and 2 Fire Upgrades (Restart) - DCP 3270
Development and Implementation of Whole Site Probabilistic Safety Assessment
Bruce A VESDA Upgrade MCR/CER
Unit 8 Fire Upgrades - DCP 3328
63732 SDS2 NOP Enhancement
Legacy Registration Project DCN/DCPs – Bruce A
Legacy Registration Project DCN/DCPs – Bruce B
M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications (Units 3 to 8)
M/34720 Addition of Relief Valves For Overpressure Protection (Units 1 to 8)
I/63472 Remote Relief Valve Position Indication (Units 1 to 8)
Implementation of Asset Management Activities
Ongoing Work on Acoustic Channels and End Plate Cracking

10. Development of GIOs

Collections of micro-gaps grouped under GIs that were developed in Section 8.2.5 and other safety related improvement initiatives identified in Section 9 were integrated as appropriate and defined as GIOs. This step resulted in the set of 51 GIOs:

- 12 new GIOs for the micro-gaps identified in Section 8.2.5.
- 16 GIOs for the MCR initiatives identified in Section 9.2 as part of the MCR scope.
- 12 GIOs for the other initiatives identified in Section 9.3.
- 11 GIOs in-progress included in the 2014 IIP [23].

The integrated list of ranked GIOs, including those in the 2014 IIP [23], is provided in Section 11.

11. Prioritization and Ranking of GIOs

The purpose of this step is to arrive at a list of GIOs from Section 10 ranked in order of priority based on the magnitude and timeliness of the benefit to be achieved by solving them. Note that this ranking only indicates the importance of the GIO, but not the feasibility of the associated corrective actions subject to constraints of cost and time or intangible considerations. The latter is part of development of the IIP. The ranking and prioritization step entails the following:

- Use the GAF described in Appendix C, as implemented in the PSR database, to assign each GIO to a second tier objective in the value tree. In so doing, the Global Issue assumes the same priority as the Tier 2 objective as expressed in the weight of the objective;
    - Defence-in-depth is a prime consideration in establishing these value tree objectives, as is evident from Appendix C (Table 37), which shows the specific levels of defence-in-depth that are supported by each of the Tier 2 objectives.
- Taking into consideration the nature of potential corrective actions for the GIO use the GAF to evaluate the impact and time-to-take-effect of resolving the GIO. In so doing, a two parameter utility score is assigned to the GIO;
- Calculate a ranking number for the GIO by multiplying the assigned weight and score; and
- Arrange the GIOs based on ranking number from highest to lowest to arrive at a ranked list. GIOs that have the same final score are given the same rank.

The results of ranking and prioritization are presented in Table 32. It should be noted that GIOs are numbered sequentially as they are developed in the database. However, given the iterative nature and the time span of the Global Assessment process, GIOs are changed or added as Safety Factor Reports and other inputs, such as the MCR scope and other initiatives, become available. Moreover, when all micro-gaps under a given GIO are resolved, the GIO no longer appears, and the GIO number is not re-used. As a result some GIO numbering does not follow a sequence.

Table 32: Ranking of GIOs

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce A & B	GIO-025	Fitness for service	Perform R&D in support of fuel channel life cycle management initiatives	1
Bruce A	GIO-028	Fitness for service	Upgrade Emergency and Standby Power Supplies	1
Bruce Units 3-8	GIO-056	Fitness for service	Fuel Channel Replacement	1
Bruce Units 3-8	GIO-057	Fitness for service	Steam Generator Replacement	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce Units 3-8	GIO-058	Fitness for service	Feeder Replacement	1
Bruce Units 3-8	GIO-062	Fitness for service	PHT Pump Seal Bellows Replacement	1
Bruce Units 3&4	GIO-064	Fitness for service	Control Distribution Frame (CDF) Terminal Replacement	1
Bruce Units 3-8	GIO-070	Fitness for service	Air Operated Valves-Replacement	1
Bruce Units 3-8	GIO-071	Fitness for service	Large Motors-Refurbishment/Replacement	1
Bruce Units 3-8	GIO-076	Fitness for service	DCC Cables and WIBAs –Replacement	1
Bruce Units 3-8	GIO-077	Fitness for service	Moderator Heat Exchangers- Replacement	1
Bruce Units 3-8	GIO-078	Fitness for service	Maintenance Cooling Heat Exchanger- Replacement	1
Bruce Units 3-8	GIO-086	Fitness for service	PHT Valves-Refurbishment of 33120-MV23	1
Bruce Units 3-8	GIO-095	Fitness for service	45VDC Power Supplies-Replacement	1
Bruce Units 3-8	GIO-100	Physical design	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications	1
Bruce A & B	GIO-101	Physical design	M/34720 Relief Valves For Overpressure Protection	1
Bruce A & B	GIO-102	Physical design	I/63472 Remote Relief Valve Position Indication	1
Bruce A & B	GIO-039	Fitness for service	Equipment Reliability and Maintenance	2
Bruce Units 3-8	GIO-059	Fitness for service	Calandria and Shield Tank Assembly Major Inspection	2
Bruce Units 3-8	GIO-060	Fitness for service	Preheater Inspections	2
Bruce Units 3-8	GIO-065	Fitness for service	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection	2
Bruce Units 3-8	GIO-066	Fitness for service	Pressurizer and Supports- Internal Inspection	2
Bruce A & B	GIO-104	Fitness for service	Ongoing Work on Bruce B Heat Transport Vibration Project	2
Bruce A	GIO-034	Fitness for service	Safety System Reliability	3
Bruce Units 1&2	GIO-019	Physical design	Assess and improve seismic qualification	4
Bruce A & B	GIO-082	Environmental protection	Performance testing of nuclear air-cleaning systems	5

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce A & B	GIO-089	Safety analysis	Whole-Site Probabilistic Risk Assessment	5
Bruce A & B	GIO-009	Safety analysis	Update safety analysis to align with REGDOC-2.4.1	6
Bruce A & B	GIO-103	Fitness for service	Implementation of Asset Management Activities	7
Bruce B	GIO-099	Physical design	Install Correctly Sized Maintenance Cooling Relief Valves	8
Bruce A & B	GIO-043	Human performance management	Validation of Human Credited Actions	9
Bruce A & B	GIO-093	Radiation protection	RP equipment and instrumentation maintenance and life cycle management	10
Bruce A & B	GIO-094	Radiation protection	Effective use of the action tracking system in Radiation Protection	11
Bruce A & B	GIO-011	Operating performance	Implement enhancements to SAMG	12
Bruce A & B	GIO-001	Physical design	Improve documented design basis	13
Bruce A & B	GIO-081	Physical design	Human Factors in Design of Nuclear Power Plants	13
Bruce A & B	GIO-083	Safety analysis	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2	13
Bruce A & B	GIO-044	Emergency management and fire protection	Emergency preparedness	14
Bruce A & B	GIO-088	Management system	Improve Licencing Processes	15
Bruce A & B	GIO-002	Physical design	Implement design changes to improve severe accident response	16
Bruce A & B	GIO-026	Physical design	BA & BB New Neutronic Trips	16
Bruce Units 3&4	GIO-090	Physical design	SDS2 Enhancements	16
Bruce A	GIO-091	Physical design	Bruce A Fire Protection Upgrades to Align with CSA-N293-07	16
Bruce B	GIO-092	Physical design	Bruce B Fire Protection Upgrades to Align with CSA-N293-07	16
Bruce A	GIO-097	Physical design	Bruce A Legacy Registration- Implementation Projects	16
Bruce B	GIO-098	Physical design	Bruce B Legacy Registration- Implementation Projects	16
Bruce A & B	GIO-024	Management system	Enhanced Periodic Safety Review to Support Asset Management	17
Bruce B	GIO-003	Physical design	Assess pipe whip and jet impingement	18

Unit	GIO No.	CNSC S&C Area	GIO TITLE	Rank
Bruce B	GIO-005	Physical design	Assess cyclic loads of pressure retaining components designed per ASME III or VIII	18
Bruce A & B	GIO-036	Physical design	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment	18
Bruce A & B	GIO-037	Physical design	Document design basis for zoning and shielding	18

Part IV: Integrated Implementation Plan

Section	Title
12	Development of Corrective Actions
13	Prioritization and Ranking of Corrective Actions
14	Application of RIDM
15	Integrated Implementation Plan and Associated High Level Corrective Actions
16	Optimization of the IIP

Appendix	Title
Appendix A	Integrated Implementation Plan

12. Development of Corrective Actions

This step, as described in Section 5.7, provides for the identification and high level definition of Corrective Actions (CAs) to address each of the GIOs as described in Section 10.

Consultation with Bruce Power SMEs and stakeholders was an essential part of the development of CAs.

CAs were defined such that all micro-gaps and initiatives consolidated under each CA can be mapped under a single Tier 3 sub-objective of the value tree described in Appendix C so that their relative ranking and prioritization can be performed in a consistent manner. Appendix H includes a template for each CA and contains the list of micro-gaps addressed under each CA.

The full list of CAs, which includes those in 2014 IIP [11], are provided in Appendix A. It should be noted that all MCR related CAs are identified for Units 3-8 on a unit specific basis.

13. Prioritization and Ranking of Corrective Actions

The prioritization and ranking of CAs uses the GAF as described in Section 5.8.

At this step, CAs associated with GIO in the 2014 IIP [11] was also included in the ranking process. This provides an integrated ranking of all the CAs and associated GIOs. The results of ranking are shown in Table 33. Details of ranking including time and impact assessments are presented in Appendix H. It should be noted that all CAs and their associated GIOs in 2014 IIP [11] were included to provide a full overview of their relative ranking. The transition plan for reporting status of the IIP is addressed in Section 17.

Table 33: Listing of GIOs and CARDs (Including Rank)

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
Bruce A & B	GIO-001	Physical design	Improve documented design basis	CA-0006	SIP-13B: BB Legacy Registration	15
				CA-0191	Update governing procedures and implementing documents on seismic qualification	15
Bruce A & B	GIO-002	Physical design	Implement design changes to improve severe accident response	CA-0009	SIP-1A: Fukushima Response - Bruce A External Water Makeup to Heat Transport System and Moderator System	23
				CA-0010	SIP-1B: Fukushima Response - Bruce B External Water Makeup to Heat Transport System and Moderator System	23
				CA-0011	SIP-2A: Fukushima Response - Bruce A Containment Venting Connection Point and Passive CFVS Installation	23
				CA-0012	SIP-2B: Fukushima Response Bruce B - Containment Venting Connection Point and Passive CFVS Installation	23
				CA-0013	SIP-4: Fukushima Response (SAMG Improvement) - Bruce A Wide range ECI Sump Level Indication	23
Bruce B	GIO-003	Physical design	Assess pipe whip and jet impingement	CA-0192	SF1-3: Perform an assessment of pipe whip and jet impingement	19
Bruce B	GIO-005	Physical design	Assess cyclic loads of pressure retaining components designed per ASME III or VIII	CA-0028	SF1-8: Evaluate impact of fatigue due to cyclic operation transient loads on Class 4 Containment Penetrations	19
				CA-0029	SF1-9: Evaluate impact of fatigue for Class 2, 3 and 4 bellows expansion joints	19
				CA-0030	SF1-12: Evaluate Class 6 piping components for cyclic and dynamic reactions	19
Bruce A & B	GIO-009	Safety analysis	Update safety analysis to align with	CA-0043	SIP-3:REGDOC-2.4.1 Implementation	8

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
			REGDOC-2.4.1	CA-0174	Safety Report & Probabilistic Safety Assessment	8
Bruce A & B	GIO-011	Operating performance	Implement enhancements to SAMG	CA-0047	SIP-11: Fukushima Response - Severe Accident Management Enhancements	17
Bruce Units 1&2	GIO-019	Physical design	Assess and improve seismic qualification	CA-0061	SIP-16: BA U1/U2 Post RTS - Seismic Margin Upgrade (IIP-6)	5
Bruce A & B	GIO-024	Management system	Enhanced Periodic Safety Review to Support Asset Management	CA-0066	SIP-22: Enhanced Periodic Safety Review to Support Asset Management	24
Bruce A & B	GIO-025	Fitness for service	Perform R&D in support of fuel channel life cycle management initiatives	CA-0067	Fuel Channel Life Management	1
Bruce A & B	GIO-026	Physical design	BA & BB New Neutronic Trips	CA-0069	SIP-25: BA & BB New Neutronic Trips Feasibility Project	23
Bruce A	GIO-028	Fitness for service	Upgrade Emergency and Standby Power Supplies	CA-0071	SIP-30: BA U1/U2 Post RTS - Standby Generator Controls Replacement	1
				CA-0073	SIP-35: Emergency Power Generators 1 and 2 Upgrades	1
Bruce A	GIO-034	Fitness for service	Safety System Reliability	CA-0078	Improvement of unavailability targets for some safety related systems	4
Bruce A & B	GIO-036	Physical design	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment	CA-0080	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment	19
Bruce A & B	GIO-037	Physical design	Document design basis for zoning and shielding	CA-0081	Establish technical basis for radiation zone designation	19
				CA-0082	Shielding design criteria and the methodology for specification of shielding parameters and material selection	19
Bruce A & B	GIO-039	Fitness for service	Equipment Reliability and Maintenance	CA-0084	In-Service Inspection Program for Bruce NGS A and B Safety Related Structures	2
Bruce A & B	GIO-043	Human performance management	Validation of Human Credited Actions	CA-0089	Validation of human actions credited under accident conditions in the safety report	11

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0177	Definition of staff availability requirements for supporting heat sink availability	18
Bruce A & B	GIO-044	Emergency management and fire protection	Emergency preparedness	CA-0090	Emergency response documentation	20
				CA-0199	Complete the On-Site/Off-Site Emergency Response Communications Project	21
				CA-0200	Addressing outstanding follow-up actions from Audits on Emergency Preparedness	16
				CA-0201	Address the following issues identified as part of the OSART review with respect to ERP	14
Bruce Units 3-8	GIO-056	Fitness for service	Fuel Channel Replacement	CA-0120	Fuel Channel Replacement - Unit 6	1
				CA-0209	Fuel Channel Replacement - Unit 3	1
				CA-0226	Fuel Channel Replacement - Unit 4	1
				CA-0243	Fuel Channel Replacement - Unit 5	1
				CA-0260	Fuel Channel Replacement - Unit 7	1
				CA-0277	Fuel Channel Replacement - Unit 8	1
Bruce Units 3-8	GIO-057	Fitness for service	Steam Generator Replacement	CA-0121	Steam Generator Replacement - Unit 6	1
				CA-0210	Steam Generator Replacement - Unit 3	1
				CA-0227	Steam Generator Replacement - Unit 4	1
				CA-0244	Steam Generator Replacement - Unit 5	1
				CA-0261	Steam Generator Replacement - Unit 7	1
				CA-0278	Steam Generator Replacement - Unit 8	1
Bruce Units 3-8	GIO-058	Fitness for service	Feeder Replacement	CA-0122	Feeder Replacement - Unit 6	1
				CA-0211	Feeder Replacement - Unit 3	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0228	Feeder Replacement - Unit 4	1
				CA-0245	Feeder Replacement - Unit 5	1
				CA-0262	Feeder Replacement - Unit 7	1
				CA-0279	Feeder Replacement - Unit 8	1
Bruce Units 3-8	GIO-059	Fitness for service	Calandria and Shield Tank Assembly Major Inspection	CA-0123	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 6	2
				CA-0212	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 3	2
				CA-0229	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 4	2
				CA-0246	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 5	2
				CA-0263	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 7	2
				CA-0280	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 8	2
Bruce Units 3-8	GIO-060	Fitness for service	Preheater Inspections	CA-0124	Preheater Inspections - Unit 6	2
				CA-0213	Preheater Inspections - Unit 3	2
				CA-0230	Preheater Inspections - Unit 4	2
				CA-0247	Preheater Inspections - Unit 5	2
				CA-0264	Preheater Inspections - Unit 7	2
				CA-0281	Preheater Inspections - Unit 8	2
Bruce Units 3-8	GIO-062	Fitness for service	PHT Pump Seal Bellows Replacement	CA-0126	PHT Pump Seal Bellows Replacement - Unit 6	1
				CA-0215	PHT Pump Seal Bellows Replacement - Unit 3	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0232	PHT Pump Seal Bellows Replacement - Unit 4	1
				CA-0249	PHT Pump Seal Bellows Replacement - Unit 5	1
				CA-0266	PHT Pump Seal Bellows Replacement - Unit 7	1
				CA-0283	PHT Pump Seal Bellows Replacement - Unit 8	1
Bruce Units 3&4	GIO-064	Fitness for service	Control Distribution Frame (CDF) Terminal Replacement	CA-0334	Control Distribution Frame (CDF) Terminal Replacement - Unit 3	7
				CA-0335	Control Distribution Frame (CDF) Terminal Replacement - Unit 4	7
Bruce Units 3-8	GIO-065	Fitness for service	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection	CA-0129	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection - Unit 6	2
				CA-0336	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection - Unit 3	2
				CA-0337	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection - Unit 4	2
				CA-0338	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection - Unit 5	2
				CA-0339	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection - Unit 7	2
				CA-0340	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection - Unit 8	2
Bruce Units 3-8	GIO-066	Fitness for service	Pressurizer and Supports- Internal Inspection	CA-0130	Pressurizer and Supports- Internal Inspection – Unit 6	2
				CA-0341	Pressurizer and Supports- Internal Inspection – Unit 5	2
				CA-0342	Pressurizer and Supports- Internal Inspection – Unit 7	2
				CA-0343	Pressurizer and Supports- Internal Inspection – Unit 8	2

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0344	Pressurizer and Supports- Internal Inspection – Unit 3	2
				CA-0345	Pressurizer and Supports- Internal Inspection – Unit 4	2
Bruce Units 3-8	GIO-070	Fitness for service	Air Operated Valves-Replacement	CA-0138	Air Operated Valve- Nuclear Valve Replacement - Unit 6	1
				CA-0217	Air Operated Valve- Nuclear Valve Replacement - Unit 3	1
				CA-0234	Air Operated Valve- Nuclear Valve Replacement - Unit 4	1
				CA-0251	Air Operated Valve- Nuclear Valve Replacement - Unit 5	1
				CA-0268	Air Operated Valve- Nuclear Valve Replacement - Unit 7	1
				CA-0285	Air Operated Valve- Nuclear Valve Replacement - Unit 8	1
				CA-0139	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 6	1
				CA-0329	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 3	1
				CA-0330	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 4	1
				CA-0331	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 5	1
				CA-0332	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 7	1
				CA-0333	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 8	1
Bruce Units 3-8	GIO-071	Fitness for service	Large Motors-Refurbishment/Replacement	CA-0145	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 6	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0346	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 3	1
				CA-0347	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 4	1
				CA-0348	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 5	1
				CA-0352	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 7	1
				CA-0353	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 8	1
Bruce Units 3-8	GIO-076	Fitness for service	DCC Cables and WIBAs –Replacement	CA-0153	DCC Cables and WIBAs -Replacement - Unit 6	1
				CA-0221	DCC Cables and WIBAs -Replacement - Unit 3	1
				CA-0238	DCC Cables and WIBAs -Replacement - Unit 4	1
				CA-0255	DCC Cables and WIBAs -Replacement - Unit 5	1
				CA-0272	DCC Cables and WIBAs -Replacement - Unit 7	1
				CA-0289	DCC Cables and WIBAs -Replacement - Unit 8	1
Bruce Units 3-8	GIO-077	Fitness for service	Moderator Heat Exchangers-Replacement	CA-0154	Moderator Heat Exchangers- Replacement - Unit 6	1
				CA-0222	Moderator Heat Exchangers- Replacement - Unit 3	1
				CA-0239	Moderator Heat Exchangers- Replacement - Unit 4	1
				CA-0256	Moderator Heat Exchangers- Replacement - Unit 5	1
				CA-0273	Moderator Heat Exchangers- Replacement - Unit 7	1
				CA-0290	Moderator Heat Exchangers- Replacement - Unit 8	1
Bruce Units 3-8	GIO-078	Fitness for service	Maintenance Cooling Heat Exchanger- Replacement	CA-0155	Maintenance Cooling Heat Exchanger- Replacement - Unit 6	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0223	Maintenance Cooling Heat Exchanger- Replacement - Unit 3	1
				CA-0240	Maintenance Cooling Heat Exchanger- Replacement - Unit 4	1
				CA-0257	Maintenance Cooling Heat Exchanger- Replacement - Unit 5	1
				CA-0274	Maintenance Cooling Heat Exchanger- Replacement - Unit 7	1
				CA-0291	Maintenance Cooling Heat Exchanger- Replacement - Unit 8	1
Bruce A & B	GIO-081	Physical design	Human Factors in Design of Nuclear Power Plants	CA-0179	HF design review of control room, workstations, computer interfaces, alarms systems, soft control systems, communication systems and field components relevant to safety	19
Bruce A & B	GIO-082	Environmental protection	Performance testing of nuclear air-cleaning systems	CA-0168	Air pressure measurements in support of emission estimates	4
				CA-0169	QA/QC guidance for performance testing of nuclear air-cleaning systems	4
				CA-0170	Effectiveness reviews of the air-cleaning system performance testing program	4
				CA-0171	Requirements for the qualifications of personnel who conduct air filter performance testing	4
				CA-0172	Performance testing of nuclear air-cleaning systems- Program documentation	4
				CA-0173	Pre-service and in-service testing of adsorbent media (activated carbon)	4
Bruce A & B	GIO-083	Safety analysis	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2	CA-0190	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2	22
Bruce Units	GIO-086	Fitness for service	PHT Valves-Refurbishment of 33120-	CA-0207	PHT Valves-Refurbishment of 33120-MV23 - Unit 6	2

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
3-8			MV23	CA-0224	PHT Valves-Refurbishment of 33120-MV23 - Unit 3	2
				CA-0241	PHT Valves-Refurbishment of 33120-MV23 - Unit 4	2
				CA-0258	PHT Valves-Refurbishment of 33120-MV23 - Unit 5	2
				CA-0275	PHT Valves-Refurbishment of 33120-MV23 - Unit 7	2
				CA-0292	PHT Valves-Refurbishment of 33120-MV23 - Unit 8	2
Bruce A & B	GIO-088	Management system	Improve Licencing Processes	CA-0294	Licence Concessions Database	3
Bruce A & B	GIO-089	Safety analysis	Whole-Site Probabilistic Risk Assessment	CA-0376	Development and Implementation of Whole-Site Probabilistic Risk Assessment	6
Bruce Units 3&4	GIO-090	Physical design	SDS2 Enhancements	CA-0297	Implement SDS2 Neutron Overpower Protection Enhancements - Unit 3	23
				CA-0378	Implement SDS2 Neutron Overpower Protection Enhancements - Unit 4	23
Bruce A	GIO-091	Physical design	Bruce A Fire Protection Upgrades to Align with CSA-N293-07	CA-0299	BA ASB Fire Protection Upgrades	23
				CA-0300	Unit 1 and 2 Fire Upgrades (Restart - Project #38730)	23
				CA-0301	BA Standby Generator Building Fire Protection Upgrade	23
				CA-0302	Bruce A Fire Barriers Upgrades (Cable Wraps)	23
				CA-0303	Bruce A Very Early Smoke Detection Apparatus (VESDA) Upgrade	23
				CA-0304	Unit 1 and 2 Fire Upgrades (SCA VESDA & Turbine Sprinkler System alarm detection and notification interface)	23
Bruce B	GIO-092	Physical design	Bruce B Fire Protection Upgrades to Align with CSA-N293-07	CA-0306	BB U0 Fuel Storage Area Sprinkler Upgrades	23
				CA-0307	Bruce B Fireworks Terminal Replacement	23

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0308	Bruce B Firewater Pipe Replacement	23
				CA-0309	Bruce B Fire Detection Upgrade	23
				CA-0310	Bruce B Very Early Smoke Detection Apparatus (VESDA) Upgrade	23
				CA-0311	Bruce B Fire Barriers (Cable Wrap) upgrades	23
				CA-0312	Bruce B Standby Generator Building Fire Protection Upgrade	23
				CA-0313	BB EPG / EWPS Building Fire Protection Upgrade	23
				CA-0315	Unit 8 Fire Upgrades	23
				CA-0316	Air Foam System Replacement	23
Bruce A & B	GIO-093	Radiation protection	RP equipment and instrumentation maintenance and life cycle management	CA-0317	RP Instrumentation life cycle management	13
				CA-0318	RP Instrumentation maintenance	12
				CA-0320	Technical Basis for RP instrumentation setpoints, locations and function checks	13
Bruce A & B	GIO-094	Radiation protection	Effective use of the action tracking system in Radiation Protection	CA-0319	Improve effective use of the action tracking system in Radiation Protection	13
Bruce Units 3-8	GIO-095	Fitness for service	45VDC Power Supplies-Replacement	CA-0321	45VDC Power Supplies-Replacement - Unit 0A	2
				CA-0322	45VDC Power Supplies-Replacement - Unit 0B	2
				CA-0323	45VDC Power Supplies-Replacement - Unit 3	2
				CA-0324	45VDC Power Supplies-Replacement - Unit 4	2
				CA-0325	45VDC Power Supplies-Replacement - Unit 5	2
				CA-0326	45VDC Power Supplies-Replacement - Unit 6	2
				CA-0327	45VDC Power Supplies-Replacement - Unit 7	2

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0328	45VDC Power Supplies-Replacement - Unit 8	2
Bruce A	GIO-097	Physical design	Bruce A Legacy Registration-Implementation Projects	CA-0298	Documentation - Legacy Registration Project DCN/DCPs- Bruce A	25
				CA-0349	Implementation - Legacy Registration Project DCN/DCPs- Bruce A	23
Bruce B	GIO-098	Physical design	Bruce B Legacy Registration-Implementation Projects	CA-0351	Implementation - Legacy Registration Project DCN/DCPs- Bruce B	23
Bruce B	GIO-099	Physical design	Install Correctly Sized Maintenance Cooling Relief Valves	CA-0314	BB Maintenance Cooling Interspace Protection	10
Bruce Units 3-8	GIO-100	Physical design	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications	CA-0354	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 3	1
				CA-0355	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 4	1
				CA-0356	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 5	1
				CA-0357	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 6	1
				CA-0358	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 7	1
				CA-0359	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 8	1
Bruce A & B	GIO-101	Physical design	M/34720 Relief Valves For Overpressure Protection	CA-0360	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 1	1
				CA-0361	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 2	1
				CA-0362	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 3	1
				CA-0363	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 4	1

Unit	GIO No.	CNSC S&C Area	GIO TITLE	CARD #	CARD TITLE	Rank
				CA-0364	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 5	1
				CA-0365	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 6	1
				CA-0366	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 7	1
				CA-0367	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 8	1
Bruce A & B	GIO-102	Physical design	I/63472 Remote Relief Valve Position Indication	CA-0368	I/63472 Remote Relief Valve Position Indication - Unit 1	1
				CA-0369	I/63472 Remote Relief Valve Position Indication - Unit 2	1
				CA-0370	I/63472 Remote Relief Valve Position Indication - Unit 3	1
				CA-0371	I/63472 Remote Relief Valve Position Indication - Unit 4	1
				CA-0372	I/63472 Remote Relief Valve Position Indication - Unit 5	1
				CA-0373	I/63472 Remote Relief Valve Position Indication - Unit 6	1
				CA-0374	I/63472 Remote Relief Valve Position Indication - Unit 7	1
				CA-0375	I/63472 Remote Relief Valve Position Indication - Unit 8	1
Bruce A & B	GIO-103	Fitness for service	Implementation of Asset Management Activities	CA-0377	Implementation of Asset Management Activities for Safety Significant Assets	9
Bruce A & B	GIO-104	Fitness for service	Ongoing Work on Bruce B Heat Transport Vibration Project	CA-0379	Bruce B Heat Transport Vibration Project	2

14. Application of RIDM

The need to perform a RIDM assessment was established in consultation with Bruce Power and based on the scope, schedule and cost considerations associated with each CA. For example, a RIDM assessment would have been required in cases where:

- The associated costs are so extensive that implementation of similar or higher ranked CAs may be delayed; or
- Other considerations such as Bruce Power's asset management plan expectations.

At this time, there were no CAs identified by Bruce Power that needed to be evaluated using B-REP-03611-00004 Risk Informed Decision Making Process.

Therefore, Table 33 in Section 13 constitutes the final list of practicable Corrective Actions that serve as input to the IIP at this time.

15. Integrated Implementation Plan and Associated High Level Corrective Action Plans

The development of the IIP entailed the following steps:

- Develop a High Level Corrective Action Plan (CAP) for Each Corrective Action;
- Optimize the IIP; and
- Prepare IIP

Results of each of these steps are addressed in the following sections.

15.1. Develop a High Level Plan for Corrective Actions

CAs are designated as CARDs (Corrective Action Requirement Descriptions) in the PSR database. CA descriptions are included in Appendix H define at a high level associated requirements for implementation. These descriptions were considered to be sufficient for initiation of implementation actions by Bruce Power. The details of the implementation actions may not be known at the time of IIP approval. These details will be communicated in future regular updates on the status of the CAs. On the other hand, some CAs may be existing Bruce Power actions that contain much greater detail on actions being taken.

In order to minimize potential duplication and the effort associated with preparing CAPs, Project Plans or Action Tracking ARs or similar documentation that are already in place were identified for each CA and listed in Appendix H. Additional CAPs will be prepared based on input from Bruce Power on an as required basis.

15.2. High Level Scope

Each CARD defines the high level scope for their respective GIO. Based on input from Bruce Power, CARDs integrate the improvements identified and related projects, planned actions to close the related CNSC AIs, planned maintenance, inspections, and any other activities. Appropriate links to the relevant Project Plans, Bruce Power Action Tracking System ARs, Regulatory commitments, etc., are listed in Appendix H. It should be noted that some CARDs include milestone dates, which indicates that more information is required before the mitigating action can be implemented. Furthermore, any CARDs that include a milestone date will remain open until the mitigating action has been completed.

15.3. High Level Schedule

A high level schedule for each CA is included in Appendix A and any further details are provided in Appendix H for each CA. The high level schedule will be updated as part of the IIP progress reports to the CNSC.

16. Optimization of the IIP

The purpose of this step is to determine the optimal feasible sequence for implementing high priority corrective actions subject to the limitations imposed by scope, schedule, cost, outage length and frequency, resource availability and other constraints. An important consideration of this step is to review the relationships between corrective actions irrespective of their ranking and based on implementation effectiveness. Those corrective actions or their elements which may be a pre-requisite to another or those where their implementation and timing present economies of scale would be planned accordingly.

Specifically, an integrated review with the MCR plans, other asset management initiatives and associated corrective actions will be performed periodically to remove potential duplication, identify opportunities for optimization of scope, resource needs and schedule.

Based on input from Bruce Power, there was no need for further integration based on the review of ranked and prioritized CAs that would optimize available resources and time and to maximize the safety benefit. In Appendix H unit or station specific initiatives are specified accordingly for each CA.

IIP is consolidated with those improvement initiatives documented in [11] and tabulated in Appendix A including CNSC Safety and Control Area that is primarily applicable to each GIO.

An IIP was prepared and submitted to the CNSC in support of operation of Bruce A and Bruce B for the current licence period on October 31, 2014. The 2014 IIP was an output of the Bruce A and Bruce B Safety Basis Report [31], which is essentially the same as the PSR process followed in this report. In this context, improvement initiatives documented in [11] are an integral part of the IIP documented in this report.

The first column in Appendix A refers to two sets of initiatives designated by the year they were developed. Those initiatives designated as 2014 were documented in [11] in support of the licence renewal for the current period. Initiatives designated as 2015 or 2016 are those developed during this PSR process.

17. IIP Progress Monitoring, Change Control and Updates

The IIP comprises projects and other activities such as CNSC Action Items (AIs) that are managed through current Bruce Power governance and the implementing programs and procedures. In addition, Bruce Power has developed BP-PROC-01024, Periodic Safety Reviews, to implement the PSR process. The individual elements of the IIP will continue to be managed through these applicable processes, including the requisite management oversight, change control and regulatory reporting.

The existing processes that include progress monitoring and change control are:

- Project Progress Monitoring
    - BP-PROC-00041, Contract Management
- Change Control
    - Manage and Control Project Change, BP-PROC-14406
    - Monitor and Control Project Scope, BP-PROC-14420
    - Design Change Package, BP-PROC-00539
- Regulatory Interface
    - CNSC Commitment Management, BP-PROC-00058
    - Evaluation of Proposed Changes for CNSC Review and Approval, BP-PROC-00090
    - BP-PROC-00067, CNSC - Bruce Power Interface Protocols
    - Formal Correspondence with the CNSC, BP-PROC-00064

These processes include requirements to assess the impact on safety of proposed changes to a project. Therefore, each individual element of the IIP is controlled to ensure that changes to it have no significant adverse impact on safety. Otherwise, justification is provided or mitigating measures put in place.

Progress on the IIP as an integrated entity will be monitored as part of the PSR process. Each update of progress on the IIP, as well as changes to the IIP, such as delays, new scope, etc., will be evaluated in terms of its impact on plant safety. Improvements realized by complementing of IIP initiatives will also be fed back to sustaining programs for on-going plant operations, PSR and Asset Life Management Options to sustain plant safety and reliability and achieve continuous improvement. In relation to monitoring and change control of the IIP, updates will include the following:

- The status of each element of the IIP, identification of approved changes in scope and/or schedule for activities within the IIP, and identification of new activities in the IIP approved since the previous update.
- An integrated assessment of the effect on safety of:
    - IIP activities completed since the previous Safety Basis update;
    - Partial progress on IIP activities where there is some benefit, but the activity is not fully complete; and
    - Delays, if any, in completion of IIP activities.
- Audits and self-assessments will be performed, subject to Bruce Power program auditing requirements.

As part of the transition to a single IIP, an update to the current IIP [11] was last submitted in December 2015 [23]. Further updates to that IIP will be not provided, since it will be incorporated into the single IIP presented in this report.

Part V: Justification for Proposed Continued Operations

Section	Title
18	Assessment of Defence-in-Depth
19	Assessment of Overall Safety
20	Justification for Proposed Continued Operations

Appendix	Title

No supporting Appendices required.

18. Assessment of Defence-in-Depth

18.1. Methodology for Assessment of Defence-in-Depth

The purpose of this assessment is to address the extent to which the safety requirements of defence-in-depth are fulfilled at Bruce A and Bruce B. The assessment was performed in accordance with the guidance provided in Section 5.11.1.

18.2. Definition of Safety Principles Used for Assessment of Defence-in-Depth

Based on Table 2 of SRS-No.46 Assessment of Defence in Depth for Nuclear Power Plants [29], there are a total of 53 safety principles that were considered for the assessment. These principles are grouped under the following main topics:

- Siting (S)
- Design (D)
- Manufacture and Construction (M&C)
- Commissioning (C)
- Operation (O)
- Accident Management (AM)
- Emergency Preparedness (EP)

These principles are listed in Table 34, together with the applicable levels of Defence-in-Depth (DID) (see Table 39 in Appendix C for a description of DID Levels), SRS-46 Objective Tree figure numbers pertaining to each, as well as Safety Factors related to them. In establishing which Safety Factors apply to each Safety Principle in Table 34, the specific Review Tasks and Regulations, Codes and Standards to which the Safety Factors are associated are implicitly included. This is based on a detailed mapping of each Safety Principle to the Safety Factor Report Review Tasks and Regulations, Codes and Standards provided in Reference [34].

The table is arranged such that safety principles applicable to the greatest number of DID levels are listed at the top. Ascending order of DID levels is used as the secondary ordering sequence.

Safety principles related to each level of DID:

- Safety Principles Related to Level 1 Defence-in-Depth: 35 of 53
- Safety Principles Related to Level 2 Defence-in-Depth: 32 of 53
- Safety Principles Related to Level 3 Defence-in-Depth: 37 of 53
- Safety Principles Related to Level 4 Defence-in-Depth: 33 of 53
- Safety Principles Related to Level 5 Defence-in-Depth: 7 of 53

Table 34: Relation of Safety Principles to DID Levels, SRS-46 Objective Trees, and Safety Factors

SP No.	Safety principle (SP) (extracted from Table 2 of SRS-46)	DID Levels	SRS-46 Figure #'s for Objective Trees	SF
S-138	Radiological impact on the public and the local environment	12345	12, 13, 14, 78	1, 6, 7, 14
O-265	Organization, responsibilities and staffing	12345	62, 78	2, 10, 11, 12, 13
O-296	Engineering and technical support of operations	12345	70, 78	2, 10, 12
S-142	Ultimate heat sink provisions	1234	15	1
D-150	Design management	1234	16	1, 5, 8, 10, 12
D-154	Proven technology	1234	17	1, 2, 4, 5, 7
D-158	General basis for design	1234	18	1, 3, 4, 5, 6, 7
D-186	Inspectability of safety equipment	1234	25	1, 4
D-205	Startup, shutdown and low power operation	1234	37	1, 5
D-227	Monitoring of plant safety status	1234	47, 48	1, 3, 5, 12
D-230	Preservation of control capability	1234	49	1, 2, 4, 5, 7
M&C-246	Safety evaluation of design	1234	55	1, 5, 6, 7
M&C-249	Achievement of quality	1234	56	1, 2, 3, 4, 10, 11
C-255	Verification of design and construction	1234	57, 58	1, 4, 5, 11
C-258	Validation of operating and functional test procedures	1234	59	1, 11
C-260	Collection of baseline data	1234	60	1, 4, 11
C-262	Pre-operational adjustment of plant	1234	61	1, 4, 11
O-269	Safety review procedures	1234	63	10, 11
O-292	Radiation protection procedures	1234	69	1, 8, 10, 11, 12, 14, 15
O-299	Feedback of operating experience	1234	71	8, 9, 10, 11, 15
O-305	Maintenance, testing and inspection	1234	72	2, 3, 4, 8, 10, 11, 12, 14, 15
O-312	Quality assurance in operation	1234	73	10, 11
D-192	Protection against power transient accidents	123	27, 28, 29	1, 5
D-195	Reactor core integrity	123	30, 31, 32	1, 4, 5
O-278	Training	123	65	5, 8, 9, 10, 11, 12, 13, 15
O-284	Operational limits and conditions	123	66	1, 2, 4, 5, 10, 11, 12

SP No.	Safety principle (SP) (extracted from Table 2 of SRS-46)	DID Levels	SRS-46 Figure #'s for Objective Trees	SF
O-290	Emergency operating procedures	234	68	5, 6, 7, 10, 11, 12, 13
D-164	Plant process control systems	12	19, 20	1, 2, 4, 5, 7
D-203	Normal heat removal	12	35. 36	1, 4
D-209	Reactor coolant system integrity	12	40, 41	1, 4, 5, 7
D-240	New and spent fuel storage	12	52	1, 2, 4
D-242	Physical protection of plant	12	53, 54	1, 7, 13, 14
D-200	Automatic shutdown systems	34	33, 34	1, 4, 5, 6
D-207	Emergency heat removal	34	38, 39	1
D-217	Confinement of radioactive material	34	42, 43, 44	1, 4
D-221	Protection of confinement structure	34	45, 46	1, 4, 5, 6
D-233	Station blackout	34	50	1, 5
EP-333	Emergency plans	45	77, 78	7, 10, 11, 12, 13, 15
EP-336	Emergency response facilities	45	77, 78	1, 11, 13
S-136	External factors affecting the plant	1	11	1, 6, 7, 14
D-188	Radiation protection in design	1	26	1, 5, 15
O-272	Conduct of operations	1	64	1, 10, 11, 12
O-288	Normal operating procedures	1	67	10, 11, 12
D-168	Automatic safety systems	3	21	1, 2, 4, 5, 6, 7, 8
D-174	Reliability targets	3	22	1, 2, 4, 5, 6, 8, 10
D-177	Dependent failures	3	23	1, 3, 5, 7
D-182	Equipment qualification	3	24	1, 2, 3, 4, 5, 7
D-237	Control of accidents within the design basis	3	51	1, 5, 11, 12
AM-318	Strategy for accident management	4	74	1, 5, 6, 11, 13
AM-323	Training and procedures for accident management	4	75	1, 5, 10, 11, 12, 13
AM-326	Engineered features for accident management	4	76	1, 5, 13
S-140	Feasibility of emergency plans	5	78	1, 7, 13, 14
EP-339	Assessment of accident consequences and radiological monitoring	5	78	5, 11, 13, 15

18.3. Integrated Review of Defence-in-Depth Including Results from SFRs, GA and IIP Based on Safety Principles

Results of the DID assessment of each safety principle are discussed under a grouping based on the levels and combination of levels of DID as shown in Table 34.

- 3 Safety principles: DID Levels 1, 2, 3, 4 and 5 (see Section 18.3.1)
- 19 Safety principles: DID Levels 1, 2, 3 and 4 (see Section 18.3.2)
- 4 Safety principles: DID Levels 1, 2 and 3 (see Section 18.3.3)
- 1 Safety principle: DID Levels 2, 3 and 4 (see Section 18.3.4)
- 5 Safety principles: DID Levels 1 and 2 (see Section 18.3.5)
- 5 Safety principles: DID Levels 3 and 4 (see Section 18.3.6)
- 2 Safety principles: DID Levels 4 and 5 (see Section 18.3.7)
- 4 Safety principles: DID Level 1 (see Section 18.3.8)
- 5 Safety principles: DID Level 3 (see Section 18.3.9)
- 3 Safety principles: DID Level 4 (see Section 18.3.10)
- 2 Safety principles: DID Level 5 (see Section 18.3.11)

Each review is applicable to Bruce A and B unless there is specific reference made to one station.

A review of each safety principle was performed to establish the contribution of the proposed IIP to the various DID levels. Each review is structured according to the breakdown described in Section 5.11.1 (Point 4). Table 35 summarizes the results at the GIO level. These results are also tabulated in the review of each safety principle. It is noted that GIOs with ID designations below -034 are those included in the IIP developed in 2014 [11]. The remaining GIOs are those identified in the Bruce A and Bruce B PSR and the resulting IIP. As will be shown in the remaining sub-sections of Section 18.3, there are effective and adequate provisions in place for the levels of defence-in-depth applicable to each safety principle. In some cases, GIOs were identified that would further improve the provisions for specific safety principles.

Table 35: Relation of Safety Principles to DID Levels and GIOs

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
S-138	Radiological impact on the public and the local environment	1234	GIO-002 Implement design changes to improve severe accident response
S-142	Ultimate heat sink provisions	1234	GIO-002 Implement design changes to improve severe accident response
D-150	Design management	1234	GIO-081 Human Factors in Design of Nuclear Power Plants

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
D-158	General basis for design	1234	GIO-001 Improve documented design basis
			GIO-003 Assess pipe whip and jet impingement
			GIO-005 Assess cyclic loads of pressure retaining components designed per ASME III or VIII
			GIO-009 Update safety analysis to align with REGDOC-2.4.1
			GIO-024 Enhanced Periodic Safety Review to Support Asset Management
			GIO-083 Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
			GIO-089 Whole-Site Probabilistic Risk Assessment
			GIO-097 Bruce A Legacy Registration-Implementation Projects
			GIO-098 Bruce B Legacy Registration-Implementation Projects
			GIO-100 M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
			GIO-101 M/34720 Relief Valves For Overpressure Protection
			GIO-102 I/63472 Remote Relief Valve Position Indication
			GIO-103 Implementation of Asset Management Activities
D-186	Inspectability of safety equipment	1234	GIO-059 Calandria and Shield Tank Assembly Major Inspection
			GIO-060 Preheater Inspections
			GIO-062 PHT Pump Seal Bellows Replacement
			GIO-065 PHT Seismic Restraints (Snubbers)- Periodic Inspection Program (PIP)- Inspection
			GIO-066 Pressurizer and Supports- Internal Inspection
D-205	Startup, shutdown and low power operation	1234	GIO-076 DCC Cables and WIBAs –Replacement
			GIO-077 Moderator Heat Exchangers- Replacement
			GIO-078 Maintenance Cooling Heat Exchanger- Replacement

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
			GIO-086 PHT Valves-Refurbishment of 33120-MV23
			GIO-090 SDS2 Enhancements
			GIO-099 Install Correctly Sized Maintenance Cooling Relief Valves
			GIO-100 M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
			GIO-101 M/34720 Relief Valves For Overpressure Protection
			GIO-102 I/63472 Remote Relief Valve Position Indication
D-227	Monitoring of plant safety status	1234	GIO-076 DCC Cables and WIBAs –Replacement
D-230	Preservation of control capability	1234	GIO-090 SDS2 Enhancements
			GIO-095 45VDC Power Supplies-Replacement
M&C-246	Safety evaluation of design	1234	GIO-009 Update safety analysis to align with REGDOC-2.4.1
			GIO-083 Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
			GIO-089 Whole-Site Probabilistic Risk Assessment
M&C-249	Achievement of quality	1234	GIO-001 Improve documented design basis
			GIO-097 Bruce A Legacy Registration-Implementation Projects
			GIO-098 Bruce B Legacy Registration-Implementation Projects
			GIO-100 M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
			GIO-101 M/34720 Relief Valves For Overpressure Protection
			GIO-102 I/63472 Remote Relief Valve Position Indication
			GIO-103 Implementation of Asset Management Activities
C-255	Verification of design and construction	1234	GIO-001 Improve documented design basis
			GIO-097 Bruce A Legacy Registration-Implementation Projects
			GIO-098 Bruce B Legacy Registration-Implementation Projects

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
			GIO-100 M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
			GIO-101 M/34720 Relief Valves For Overpressure Protection
			GIO-102 I/63472 Remote Relief Valve Position Indication
			GIO-103 Implementation of Asset Management Activities
O-292	Radiation protection procedures	1234	GIO-082 Performance testing of nuclear air-cleaning systems
			GIO-093 RP equipment and instrumentation maintenance and life cycle management
			GIO-094 Effective use of the action tracking system in Radiation Protection
O-305	Maintenance, testing and inspection	1234	GIO-025 Perform R&D in support of fuel channel life cycle management initiatives
			GIO-034 Safety System Reliability
			GIO-039 Equipment Reliability and Maintenance
			GIO-056 Fuel Channel Replacement
			GIO-057 Steam Generator Replacement
			GIO-058 Feeder Replacement
			GIO-059 Calandria and Shield Tank Assembly Major Inspection
			GIO-060 Preheater Inspections
			GIO-062 PHT Pump Seal Bellows Replacement
			GIO-064 Control Distribution Frame (CDF) Terminal Replacement
			GIO-065 PHT Seismic Restraints (Snubbers)- Periodic Inspection Program (PIP)- Inspection
			GIO-066 Pressurizer and Supports- Internal Inspection
			GIO-070 Air Operated Valves-Replacement
			GIO-071 Large Motors- Refurbishment/Replacement
			GIO-076 DCC Cables and WIBAs –Replacement
			GIO-077 Moderator Heat Exchangers- Replacement

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
			GIO-078 Maintenance Cooling Heat Exchanger-Replacement
			GIO-086 PHT Valves-Refurbishment of 33120-MV23
			GIO-093 RP equipment and instrumentation maintenance and life cycle management
			GIO-095 45VDC Power Supplies-Replacement
			GIO-103 Implementation of Asset Management Activities
			GIO-104 Ongoing Work on Bruce B Heat Transport Vibration Project
O-312	Quality assurance in operation	1234	GIO-088 Improve Licencing Processes
			GIO-094 Effective use of the action tracking system in Radiation Protection
D-192	Protection against power transient accidents	123	GIO-026 BA & BB New Neutronic Trips
			GIO-036 Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
			GIO-076 DCC Cables and WIBAs –Replacement
D-195	Reactor core integrity	123	GIO-056 Fuel Channel Replacement
			GIO-059 Calandria and Shield Tank Assembly Major Inspection
O-278	Training	123	GIO-081 Human Factors in Design of Nuclear Power Plants
O-290	Emergency operating procedures	234	GIO-043 Validation of Human Credited Actions
D-209	Reactor coolant system integrity	12	GIO-056 Fuel Channel Replacement
			GIO-057 Steam Generator Replacement
			GIO-058 Feeder Replacement
			GIO-060 Preheater Inspections
			GIO-065 PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection
			GIO-066 Pressurizer and Supports- Internal Inspection
			GIO-070 Air Operated Valves-Replacement
			GIO-078 Maintenance Cooling Heat Exchanger-Replacement
			GIO-086 PHT Valves-Refurbishment of 33120-MV23

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
			GIO-099 Install Correctly Sized Maintenance Cooling Relief Valves
			GIO-100 M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
			GIO-101 M/34720 Relief Valves For Overpressure Protection
			GIO-102 I/63472 Remote Relief Valve Position Indication
D-200	Automatic shutdown systems	34	GIO-026 BA & BB New Neutronic Trips
			GIO-076 DCC Cables and WIBAs –Replacement
			GIO-090 SDS2 Enhancements
			GIO-095 45VDC Power Supplies-Replacement
D-207	Emergency heat removal	34	GIO-002 Implement design changes to improve severe accident response
			GIO-070 Air Operated Valves-Replacement
D-217	Confinement of radioactive material	34	GIO-002 Implement design changes to improve severe accident response
D-221	Protection of confinement structure	34	GIO-002 Implement design changes to improve severe accident response
EP-333	Emergency plans	45	GIO-044 Emergency preparedness
			GIO-089 Whole-Site Probabilistic Risk Assessment
S-136	External factors affecting the plant		GIO-009 Update safety analysis to align with REGDOC-2.4.1
			GIO-083 Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
			GIO-089 Whole-Site Probabilistic Risk Assessment
D-188	Radiation protection in design	1	GIO-037 Document design basis for zoning and shielding
			GIO-082 Performance testing of nuclear air-cleaning systems
O-288	Normal operating procedures	1	GIO-043 Validation of Human Credited Actions
D-168	Automatic safety systems	3	GIO-026 BA & BB New Neutronic Trips
			GIO-028 Upgrade Emergency and Standby Power Supplies
			GIO-034 Safety System Reliability

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
			GIO-036 Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
			GIO-070 Air Operated Valves-Replacement
			GIO-076 DCC Cables and WIBAs –Replacement
			GIO-090 SDS2 Enhancements
D-174	Reliability targets	3	GIO-028 Upgrade Emergency and Standby Power Supplies
			GIO-034 Safety System Reliability
			GIO-070 Air Operated Valves-Replacement
			GIO-090 SDS2 Enhancements
			GIO-095 45VDC Power Supplies-Replacement
D-177	Dependent failures	3	GIO-003 Assess pipe whip and jet impingement
			GIO-019 Assess and improve seismic margins
			GIO-091 Bruce A Fire Protection Upgrades to Align with CSA-N293-07
			GIO-092 Bruce B Fire Protection Upgrades to Align with CSA-N293-07
D-182	Equipment qualification	3	GIO-003 Assess pipe whip and jet impingement
			GIO-019 Assess and improve seismic margins
			GIO-036 Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
D-237	Control of accidents within the design basis	3	GIO-002 Implement design changes to improve severe accident response
			GIO-026 BA & BB New Neutronic Trips
			GIO-028 Upgrade Emergency and Standby Power Supplies
			GIO-034 Safety System Reliability
			GIO-036 Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
			GIO-070 Air Operated Valves-Replacement
			GIO-076 DCC Cables and WIBAs –Replacement
			GIO-090 SDS2 Enhancements
			GIO-100 M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications

SP No.	Safety principle (SP) (from INSAG-12)	DID Level	GI TITLE
			GIO-101 M/34720 Relief Valves For Overpressure Protection
			GIO-102 I/63472 Remote Relief Valve Position Indication
AM-318	Strategy for accident management	4	GIO-011 Implement enhancements to SAMG
			GIO-089 Whole-Site Probabilistic Risk Assessment
AM-326	Engineered features for accident management	4	GIO-002 Implement design changes to improve severe accident response
S-140	Feasibility of emergency plans	5	GIO-044 Emergency preparedness

A review of the strengths identified in the SFRs was completed to identify which safety principles were impacted. It should be noted that the strengths identified are applicable to both Bruce A and Bruce B throughout this assessment. A summary of the results is listed in Table 36. These strengths are also tabulated under each safety principle as applicable.

Table 36: Summary of Strengths Identified in SFRs and Relevant Safety Principles

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
SF-02-S1	The conditions of the U014 and U058 SSCs are now tracked in SHRs. Bruce Power continues to improve and streamline the SHR processes as part of ageing and asset management, integrating these improvements with their anticipated obsolescence, testing, inspection and maintenance programs.	D-227	Monitoring of plant safety status	1234
		O-305	Maintenance, testing and inspection	
SF-02-S2	Bruce Power's preventive maintenance implementation is a station priority. The station management team monitors implementation and leaders enforce accountability	D-227	Monitoring of plant safety status	1234
		O-305	Maintenance, testing and inspection	
		O-272	Conduct of operations	
SF-03-S1	The quality of the programmatic documents (i.e., programs and procedures) for the equipment qualification process is very good, with interfaces with other station procedures well identified, recent revisions and updating for most procedures, and incorporation of issues identified in audits and	D-150	Design management	1234
		D-182	Equipment qualification	3

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
	self-assessments.			
SF-03-S2	The IAEA OSART review of Bruce B completed in 2015 reviewed all aspects of the environmental qualification program and recognized its overall implementation as “good performance”. Therefore, the management of the EQ program is considered to be a strength in this report.	D-150	Design management	1234
		D-182	Equipment qualification	3
SF-04-S1	Information from the Asset Management Program is proactively used to inform the business of the future needs related to ageing and to ensure the funding and priorities can be proactively established as required to ensure effective ageing management and plant safety.	O-272	Conduct of operations	1
		O-305	Maintenance, testing and inspection	1234
SF-04-S2	Bruce B is an industry leader in the area of managing obsolescence of technology as evidenced by being awarded a WANO Strength and being the subject of a WANO Good Practice publication	O-272	Conduct of operations	1
		O-305	Maintenance, testing and inspection	1234
SF-05-S1	Bruce Power has established an integrated strategy to improve the deterministic safety analysis contained in the Safety Reports as part of its objective to reach compliance with CNSC REGDOC-2.4.1 to the maximum practicable extent over a defined transition period. Bruce Power DSA procedures have been revised in consideration of CNSC REGDOC-2.4.1 requirements and the industry Principles and Guidelines for DSA. Industry guidelines for Limit of the Operating Envelope (LOE)/Realistic Operating Envelope (ROE) and Best Estimate Analysis and Uncertainty (BEAU) methodologies are established. Moreover, “Derived Acceptance Criteria for Deterministic Safety Analysis” is issued as COG 13-9035. Bruce Power is leading or actively participating in all SRI activities.	D-158	General basis for design	1234
SF-05-S2	Bruce Power has implemented significant preventive and mitigating design modifications that are intended to provide further defence in depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.	D-192	Protection against power transient accidents	123
		D-200	Automatic shutdown systems	34

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
		D-207	Emergency heat removal	
		D-217	Confinement of radioactive material	
		D-221	Protection of confinement structure	
		D-233	Station blackout	
		D-237	Control of accidents within the design basis	3
		AM-318	Strategy for accident management	4
		AM-326	Engineered features for accident management	
SF-06-S1	Bruce Power has developed and implemented a process of continuous maintenance of the PRA model to ensure that the model is representative of the actual plant configuration and operation and testing at the station. This exceeds the requirement of CNSC REGDOC-2.4.2 (Clause 4.4) that the PRA models be updated every five years.	D-158	General basis for design	1234
SF-08-S1	A strength involves the commitment to improvements that are systematically being undertaken, based on the strong direction and guidance from NORA, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently (Same strength observed as in SF-10-S2 and SF-11-S2)	O-269	Safety review procedures	1234
		O-272	Conduct of operations	1

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
SF-08-S2	Furthermore, the audit organization has a well-developed Auditor Training program which used a Systematic Approach to Training based training design. Job Task Analysis is documented for knowledge and skill elements. The training program is documented and aligned to develop proficient auditors upon completion of qualifications. Auditors are professional and meet expectations of managers for performance as qualified auditors.	O-278	Training	123
SF-08-S3	Bruce Power's organization shares Safety Performance OPEX, Compliance Reporting and Corrective Action processes as commonly-maintained programs with Bruce B, and thus observations and lessons learned at Bruce B can be used at Bruce A. Additionally, there is an opportunity to share knowledge from Bruce B by transferring managers to Bruce A and vice-versa. Thus, strengths at each station and means to see how the other Station prevents and mitigates less desirable situations are shared to increase the corporate knowledge and experience. (Same strength observed in SF-10-3).	O-299	Feedback of operating experience	1234
SF-08-S4	Bruce Power's leading role in the modification of the 37-element fuel design (37M) ensured the requirements were understood and fully incorporated, thus ensuring integration of the design and manufacturing aspects from multiple vendors who supported the project. This strength was important in ensuring the safety improvement was completed on schedule, implemented to Operation's satisfaction and as committed to the CNSC.	D-150	Design management	1234
		D-154	Proven Technology	1234
		D-158	General basis for design	1234
SF-09-S1	The review demonstrates that Bruce Power's OPEX Program and its implementation provides for adequate feedback of safety experience from nuclear power plants (both internal and external) and of the findings of research in support of continued safe and reliable operation. In addition, the review demonstrates that Bruce Power does not confine itself to utilizing OPEX from nuclear power plants only, but makes use of OPEX from any industrial process plants. Moreover, research activities are being pursued and results are used to enhance nuclear safety and equipment performance and reliability. This is regarded as a	O-299	Feedback of operating experience	1234

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
	strength in Bruce Power's OPEX Program.			
SF-10-S1	The existence of a comprehensive suite of programs and procedures that ensure the organization and administration will be controlled and well-documented in the future. Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes.	O-269	Safety review procedures	1234
		O-312	Quality assurance in operation	
SF-10-S2	The commitments to improvements that are systematically being undertaken based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently. (Same strength observed as in SF-08-S1 and SF11-S2)	See SF-08-S1		
SF-10-S2	Bruce Power's organization shares Safety Performance OPEX, Compliance Reporting and Corrective Action processes, as commonly-maintained programs with Bruce B, so observations and lessons learned at Bruce B can be used at Bruce A. Additionally, there is an opportunity to share knowledge from Bruce B by transferring managers to Bruce A and vice-versa. Thus, strengths at each station and means to see how the other Station prevents and mitigates less desirable situations are shared to increase the corporate knowledge and experience. (Same strength observed as in SF-08-3)	See SF-08-S3		
SF-11-S1	The existence of a comprehensive suite of programs and procedures that ensure procedures will be controlled and well documented in the future. Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes and revision of their procedures to meet best industry practice. This Safety Factor 11 review found that all aspects of the processes are	O-312	Quality assurance in operation	1234

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
	satisfactory.			
SF-11-S2	The commitment to improvements that are systematically being undertaken based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently. These are discussed in detail in Safety Factor 10. This strength, however, is also directly applicable to the tasks identified for this Safety Factor and its assessment of procedures. (Same strength observed as in SF-08-S1 and SF10-S2)	See SF-08-S1		
SF-13-S1	A particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.	S-140	Feasibility of emergency plans	5
		AM-318	Strategy for accident management	4
		AM-323	Training and procedures for accident management	4
		AM-326	Engineered features for accident management	4
		EP-333	Emergency plans	45
		EP-336	Emergency response facilities	45
		EP-339	Assessment of accident consequences and radiological	5

SFR Strength ID	Description of Strengths Identified	SP No.	Safety Principle (INSAG-12)	DID Level (SRS-46)
			monitoring	
SF-15-S1	Bruce Power has a mature and comprehensive radiation protection program that, by 2009, had begun to show the effects of aging and lack of maintenance. This contributed to the loss of RP controls observed during the 2009 Alpha Contamination Incident. Since that time, Bruce Power has made progress in addressing the deficiencies through RP improvement and excellence programs (see Section of Bruce A SFR 15). Bruce Power recognized that significant change was required in all areas of RP at Bruce Power, and acted on this by developing extensive RP improvement initiatives and significantly reorganizing the RP Department at each of the Bruce Power facilities. Bruce Power has since improved and leads the way in the performance indicator for Collective Radiation Exposure (CRE) in North America. This industry-leading CRE performance has been identified as a strength in performance.	O-292	Radiation protection procedures	1234

18.3.1. Safety Principles Related to Defence-in-Depth Levels 1, 2, 3, 4, 5

There are 3 safety principles (SPs) that are related to all levels of DID. Bruce A and Bruce B design and operation is aligned with all three safety principles as demonstrated below.

S-138	Radiological impact on the public and the local environment
O-265	Organization, responsibilities and staffing
O-296	Engineering and technical support of operations

S-138 Radiological impact on the public and the local environment
Principle: Sites are investigated from the standpoint of the radiological impact of the plant in normal operation and in accident conditions.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 12, 13, 14, and 78 for DID Levels 1, 2, 3&4, and 5, respectively, as described below.

Level 1

Bruce Power is committed to review, identify, and deal with any ongoing significant environmental impacts from the station.

When Bruce A and Bruce B were designed, it was recognized that various systems would be required to control emissions to the environment and waste management systems were provided. As described in Safety Principle S-136, the potential effect(s) of the plant on population, agriculture, industry, transportation, fishing and recreation have been considered for the Bruce Power site.

In addition, Bruce Power has assessed nine separate components of the environment in the EA Report prepared in support of Bruce A Refurbishment for Life Extension and Continued Operation Project in 2005:

- Radiation and Radioactivity Environment;
- Surface Water Resources;
- Aquatic Environment;
- Atmospheric Environment;
- Geology, Hydrogeology and Seismicity;
- Terrestrial Environment;
- Land Resources;
- Cultural Heritage and Aboriginal Interests; and
- Socio-economic Conditions.

The report concluded that the Project is not likely to result in any significant adverse effects on the environment, taking into account the findings of the EA studies, including the identified mitigation measures. This provides additional support for the low radiological impact of Bruce A on the public and the local environment for extended operational life of the units.

As stated in NK21-CORR-00531-13020/NK29-CORR-00531-13487, Bruce Power will also provide the following information to the CNSC in support of the MCR Project:

- An update to the environmental risk assessment that includes MCR activities at Bruce B;
- A summary report of Bruce Power's Research and Development activities to date which are continuing under the MCR project;
- A final report for the Bruce A Unit 1&2 Refurbishment for Life Extension and Continued Operations Environmental Assessment Follow-up Monitoring Program;
- Information on aboriginal engagement activities; and
- Status of Fisheries Act authorization.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Level 2

The sampling and measurement procedures for radiological monitoring are referenced in B-PROC-00076. Most of the sampling and analyses are conducted by the Bruce Power Health Physics Laboratory, which has been accredited to the analytical laboratory standard ISO/IEC 17025 by the Canadian Association for Laboratory Accreditation (CALA). The exception is environmental gamma-ray monitoring, which is performed with thermoluminescent dosimeters supplied and analyzed by the Ontario Power Generation Health Physics Laboratory. This laboratory is also accredited by CALA. According to BP-PROC-00076, the media that are being monitored for radionuclide concentrations are air, water (drinking, surface, well, precipitation, ground), agricultural plants (fruits, vegetables, grains), animal products (meat, milk, honey), fish, sediment and soil. The results are published in the annual Environmental Monitoring Program Report, and compared with historical trends.

Bruce Power has also invested considerable resources in several research and development projects related to measurement of potential impacts on Lake Huron including:

- The Bruce A Environmental Assessment follow-up activities committed as part of the Bruce A return to service. The follow-up program includes monitoring lake temperatures, as well as monitoring source water fish densities and entrainment effects on Lake Whitefish, Spottail Shiner, and Deepwater Sculpin.
- A major study, Effects of Thermal, Chemical and Radiological Emissions on Whitefish, supported by the Natural Sciences and Engineering Research Council of Canada (NSERC) examining impacts of stressors on Lake and Round Whitefish populations. This study will examine how developing fish adapt to varying levels of external stress and whether stressors alone or in combination during embryogenesis can impact juvenile fish (after hatch).
- Participation in CANDU Owners Group studies on fluctuating temperature effects on hatching success and timing for Lake and Round Whitefish.
- A collaboration project with the Saugeen Ojibway Nation examining Lake Whitefish population structure in Lake Huron, as well as entrainment effects on Lake Whitefish.
- Bruce Power's most recent updates to the Derived Release Limits (DRLs) for Bruce A and Bruce B were completed in accordance with CSA N288.1 Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities and were included in the PROL renewal applications [NK21-CORR-00531-10873/ NK29-CORR-00531-11252). These N288.1-aligned DRLs have been added to Appendix C of the current Bruce A and B PROL [NK21-CORR-00531-11715].

Bruce Power provisions for Level 2 for this safety principle are effective and adequate.

Levels 3 & 4

Bruce Power has also invested considerable resources in several research and development projects related to measurement of potential impacts on Lake Huron including:

- The Bruce A Environmental Assessment follow-up activities committed as part of the Bruce A return to service. The follow-up program includes monitoring lake temperatures, as well as monitoring source water fish densities and entrainment effects on Lake Whitefish, Spottail Shiner, and Deepwater Sculpin.
- A major study, Effects of Thermal, Chemical and Radiological Emissions on Whitefish, supported by the Natural Sciences and Engineering Research Council of Canada (NSERC) examining impacts of stressors on Lake and Round Whitefish populations. This study will examine how developing fish adapt to varying levels of external stress and whether stressors alone or in combination during embryogenesis can impact juvenile fish (after hatch).
- Participation in CANDU Owners Group studies on fluctuating temperature effects on hatching success and timing for Lake and Round Whitefish.
- A collaboration project with the Saugeen Ojibway Nation examining Lake Whitefish population structure in Lake Huron, as well as entrainment effects on Lake Whitefish.
- Bruce Power's most recent updates to the Derived Release Limits (DRLs) for Bruce A and Bruce B were completed in accordance with CSA N288.1 Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities and were included in the PROL renewal applications [NK21-CORR-00531-10873/ NK29-CORR-00531-11252). These N288.1-aligned DRLs have been added to Appendix C of the current Bruce A and B PROL [NK21-CORR-00531-11715].

Pollution prevention principles have been incorporated into Appendix A of the Bruce Power Environmental Policy documented in BP-MSM-1, Management System Manual, where it states that:

- “Bruce Power commits to … minimizing our environmental impact and prevention of pollution by minimizing emissions, preventing spills, reducing waste, and reusing or recycling our resources.”

DPT-ENV-00016, Environmental Risk Assessment - Aspect/Impact, describes the process used for identifying and ranking environmental aspects to determine which aspects are considered Significant Environmental Aspects (SEAs). Risks and compliance associated with SEAs are considered when setting environmental objectives and targets. Bruce Power maintains an EA database to assist in management of all environmental aspects, which are listed and reviewed on a regular basis.

Bruce Power provisions for Levels 3 & 4 for this safety principle are effective and adequate.

Level 5

Follow-up by Bruce Power to the Nuclear Industry’s lessons learned (including the CNSC’s Action Items) from the Fukushima-Daiichi severe accidents has resulted in significant improvement to Bruce Power’s emergency preparedness capability. These improvements, some of which are applicable to other levels of defence-in-depth, are summarized as follows:

- Physical changes to the plant to enhance accident management such as water addition tie-in points to heat transport and moderator, enhanced shield tank pressure relief, third Emergency Power Generator (EPG);
- Emergency mitigating equipment to provide additional cooling water and power supplies,
- Improved severe accident modeling capability;
- Improved assessments and assurance of instrumentation and equipment survivability and plant habitability following severe accidents;
- Improvements to severe accident management procedure to enhance response to severe accidents, including multi-unit and Irradiated Fuel Bay (IFB) events;
- Improvements in communications capability both within the site and with outside agencies;
- The addition of an off-site Emergency Management Centre and the use an Incident Management System approach to emergency response;
- Installation of off-site real time radiation monitoring instrumentation; and
- Design provisions for a containment connection, as well as the installation of a passive Containment Filtered Venting System (CFVS).

Bruce Power provisions for Level 5 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to S-138 Radiological impact on the public and the local environment

There were no strengths identified in SFRs from the standpoint of the radiological impact on the public and local environment.

There is one GIO that will further improve the radiological impact on the public and the local environment. This is the installation of a passive CFVS, which is included in GIO-002.

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, 4, and 5, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

O-265 Organization, responsibilities and staffing
Principle: The operating organization exerts full responsibility for the safe operation of a nuclear power plant through a strong organizational structure under the line authority of the plant manager. The plant manager ensures that all elements for safe plant operation are in place, including an adequate number of qualified and experienced personnel.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 62 and 78 for DID Levels 1-4, and 5, respectively, as described below.

Levels 1 – 4

BP-MSM-1 Bruce Power Management System Manual and implementing programs and procedures put in place all elements for safe plant operation, including an adequate number of qualified and experienced personnel.

BP-MSM-1 SHEET 0002, MSM - Approved Reference Chart Authorities and Responsibilities – Sheet 0002 describes the roles, responsibilities, authority and accountabilities of personnel involved in all aspects of plant operation. The roles and responsibilities of personnel are also clearly defined in the responsibilities section of each Bruce Power procedure, including a clearly identified process owner and their associated responsibilities. BP-PROG-01.02, Bruce Power Management System (BPMS) Management provides the governing processes to control and maintain the Management System.

Bruce Power ensures adequate number of qualified and experienced personnel in compliance with requirements in the PROL condition 2.2 Minimum Shift Complement and Control Room Staffing.

Organizational roles and responsibilities are also embedded in all programs and procedures of the operating organization in accordance with the BP-MSM-1 SHEET 0002 safe operation of the plant assured under line authority of the assigned plant managers by qualified and experienced personnel as prescribed in the PROL.

BP-MSM-1, Management System Manual identifies one of the responsibilities of the CEO as leading and fostering a nuclear safety culture and establishing an organization where reporting relationships, positional authority, human resources, financial resources and corporate policy support and emphasize the overriding importance of nuclear safety. BP-PROC-00892, Nuclear Safety Culture Monitoring provides the framework for Bruce Power to monitor nuclear safety culture between formal assessment activities, in particular to have mechanisms to identify and correct potential gaps in nuclear safety culture. Output from the Nuclear Safety Culture Monitoring process is recorded in forms, such as FORM-14015 R000. The Senior Leadership Team uses the form to document and rate the ten traits of Nuclear Safety Culture of their team members providing strengths, opportunities for improvement and findings during the most recent period (a minimum of three meetings are held each calendar year). These are then used by the SLT to determine subtle changes in the Safety Culture.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Level 5

The Bruce Power Nuclear Emergency Response Plan, BP-PLAN-00001-R005, describes the concepts, structures, roles and processes needed to implement and maintain Bruce Power's radiological emergency response capability. The Nuclear Emergency Response Plan (NERP) applies to all facilities within the Bruce Power Site. The NERP was developed to support response to design basis accidents that endanger the safety of personnel in the incident station, personnel on-site, members of the public and the environment, but also takes into account requirements to support a sustained response to Beyond Design Basis events, for example a Beyond Design Basis multi-unit event resulting in an extended loss of off-site power for up to 72 hours without assistance. The NERP predominantly deals with releases of radioactive materials from fixed facilities.

For those events where accident consequences indicate that the damage is beyond that for design basis accidents, the Emergency Response Organization (ERO) will activate BP-PROC-00659, Severe Accident Management (SAM) to manage the on-site response to a severe accident and thus minimize releases to the environment. This procedure interfaces with BP-PLAN-00001 in order to utilize the structures and processes contained therein. On-going reviews of changes to emergency planning, including Fukushima Action Item follow-up studies, Huron Challenge Series follow-up, and review of the licensing basis for minimum shift complement, have resulted in a number of changes to emergency planning and supporting processes, including:

- The addition of an off-site Emergency Management Centre (EMC) and the implementation of an "Incident Management System" organization structure to emergency response, including role re-alignment;
- Communications upgrades both at the new EMC and the Central Maintenance and Laundry Facility (CMLF);
- Confirmation that the minimum shift complement is adequate for the emergency plan's planning basis;
- Improvements to severe accident management procedure to enhance response to severe accidents, including multi-unit and Irradiated Fuel Bay events; and
- Development of emergency response simulation software to enhance training for greater understanding and situation awareness of event response.

The Bruce Power Nuclear Emergency Response Plan also represents a basis for controlling changes and modifications to the Bruce Power emergency preparedness capability. This plan identifies the Shift Crew emergency staffing requirements associated with conduct of plant operations identified in BP-PROG-12.01 Conduct of Plant Operations.

The Bruce Power Nuclear Emergency Response Plan is submitted to and accepted by the CNSC. This Plan has also been discussed with, agreed to, and rehearsed with the local authorities.

Bruce Power provisions for Level 5 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-265 Organization, responsibilities and staffing

There were no strengths identified in SFRs from the standpoint of organization, responsibilities and staffing.

There are no additional planned initiatives included in the IIP that will further improve organization responsibilities and staffing.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, 4, and 5, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

O-296 Engineering and technical support of operations

Principle: Engineering and technical support, competent in all disciplines important for safety, is available throughout the lifetime of the plant.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 70 and 78 for DID Levels 1-4, and 5, respectively, as described below.

Levels 1 – 4

Availability of engineering and technical support, competent in all disciplines important for safety, throughout the lifetime of the plant is ensured through implementation of BP-PROG-02.01 Worker Staffing and BP-PROG-01.04 Leadership Talent Management.

BP-PROG-02.01 Worker Staffing and its implementing procedures describe the processes and activities of recruitment, orientation, and deployment of staff that possess the competencies required for maintaining staffing levels consistent with the requisite organization structure. It applies to employees including regular, temporary, and contract employees and requires that personnel must be recruited against current organizational competencies (technical and behavioural), which are specified in approved job documents and related selection criteria.

BP-PROG-01.04 Leadership Talent Management and its implementing procedures define the approaches and responsibilities associated with the Talent Management process for managers. The program defines how leadership is defined, how managers are selected for both their leadership and technical skills, and then how managers are on-boarded, managed and developed. It defines how Bruce Power ensures a sufficient number of managers with the right leadership and technical skills are available to deliver the business plan.

BP-PROC-00221 Succession Management Procedure ensures there are capable managers to deliver on future business plans by identifying and developing successors to management positions. This procedure is supported by BP-PROC-00468 Workforce Planning Process which ensures that Bruce Power has the right people with the right skills at the right time in the right jobs. The Workforce Planning Process is accountable for delivering a 5-year workforce plan, through the annual business planning process and integrating with the recruiting function to develop hiring plans for all divisions across site.

BP-PROC-00147, Benchmarking and Conference Activities, provides requirements for identifying and documenting lessons learned from external sources to continuously improve performance by making improvements to Processes/Procedures, Training, or System/Equipment Design. Benchmarking and conference activities foster the use of diverse information sources to identify and understand performance gaps and implement corrective actions to improve performance. Bruce Power participates in a significant array of research and development activities with other organizations. Co-operative interactions related to research with CANDU Owners Group (COG), Electric Power Research Institute (EPRI), WANO, IAEA, American Society of Mechanical Engineers, INPO, CSA, NRC (National Research Council), Canadian Nuclear Society and others are well known inside Bruce Power and throughout the industry. Bruce Power performs research in conjunction with the Ontario Ministry of the Environment and Climate Change, and attends workshops to acquire OPEX (e.g., Radiological Effluents and Environmental Workshop). In summary, Bruce Power's OPEX Program and its implementation provides for adequate feedback of safety experience from nuclear power plants (both internal and external) and of the findings of research in support of continued safe and reliable operation. The research activities are being pursued and results are used to enhance nuclear safety and equipment performance and reliability.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Level 5

Emergency Response Organization Training and Qualification Description, TQD-00005, establishes the requirements for the training and qualification of individuals assigned to specific emergency response positions, as defined in BP-PLAN-00001, following a systematic approach to training methodology. Emergency Preparedness Drill and Exercises, B-PROC-00010, provides a comprehensive list of drill and exercise objectives and provides for a schedule for conducting drills and exercises such that all of the objectives are tested within a set period of time. The schedule is reviewed at least quarterly.

On-going reviews of changes to emergency planning, including Fukushima Action Item follow-up studies, Huron Challenge Series follow-up, and review of the licensing basis for minimum shift complement, have resulted in significant improvement to Bruce Power's emergency preparedness capability. These improvements are summarized as follows:

- The addition of an off-site Emergency Management Centre and the implementation of an Incident Management System organization structure to emergency response;
- Improvements in communications capability both within the site and with outside agencies;
- Instrumentation and equipment survivability studies;
- Control room and plant habitability studies; and
- Development of emergency response simulation software to enhance training for greater understanding and situation awareness of event response.

Bruce Power provisions for Level 5 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-296 Engineering and technical support of operations

There were no strengths identified in SFRs from the standpoint of engineering and technical support of operations.

There are no additional planned initiatives included in the IIP that will further improve engineering and technical support of operations.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, 4, and 5, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

18.3.2. Safety Principles Related to Defence-in-Depth Levels 1, 2, 3, 4

Nineteen safety principles are related to Levels 1, 2, 3 and 4 of DID. Bruce A and Bruce B design and operation are aligned with all the safety principles as demonstrated below.

S-142	Ultimate heat sink provisions
D-150	Design management
D-154	Proven technology
D-158	General basis for design
D-186	Inspectability of safety equipment
D-205	Startup, shutdown and low power operation
D-227	Monitoring of plant safety status
D-230	Preservation of control capability
M&C-246	Safety evaluation of design
M&C-249	Achievement of Quality
C-255	Verification of design and construction
C-258	Validation of operating and functional test procedures
C-260	Collection of baseline data
C-262	Pre-operational adjustment to the plant
O-269	Safety review procedures
O-292	Radiation protection procedures
O-299	Feedback of operating experience-
O-305	Maintenance, testing and inspection
O-312	Quality assurance in operation

S-142 Ultimate heat sink provisions
Principle: The site selected for a nuclear power plant has a reliable long term heat sink that can remove energy generated in the plant after shutdown, both immediately after shutdown and over the longer term.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 15 for DID Levels 1-4, as described below.

Levels 1 – 4

Water for all purposes is drawn from Lake Huron through a tunnel and open intake channel which is common to all four units of each station, and through individual pump houses to each unit. Pumps, mounted in separate chambers in each pump house, supply water to the circulating water system and the Low Pressure Service Water system. Screens are provided at pump intakes to remove debris. Water drawn from the lake and used for cooling is returned to the lake via discharge duct and channel.

Bruce A and Bruce B design includes various heat sinks available for normal operation and the emergency cooling system. The normal Boiler Feedwater System is backed up by the Auxiliary Boiler Feedwater system and the Emergency Boiler Cooling System at Bruce A, and Emergency Water System at Bruce B, to provide heat removal from the boilers. The Inter-Unit Feedwater tie from other operating units can also supply emergency feedwater to any unit. Power for these systems comes from the normal Class IV power backed up by Class III standby generators or, to a more limited extent, the Qualified Power Supply at Bruce A and Emergency Power supply at Bruce B. Service water to heat exchanges and other components is supplied via the Unit Low Pressure Water Service System, the High Pressure Recirculating System or the Common Service Water System.

The Seismic and Environmental Qualification Programs that have been undertaken at Bruce A and Bruce B have demonstrated that the essential parts of the existing systems are capable of meeting their environmental and seismic requirements.

Each of these systems has been designed with redundancy, diversity and reliability in accordance with their importance to the function of heat removal. In summary, heat removal from the core is provided by a variety of systems (e.g., steam reject from the steam generators with feed water supplied by the auxiliary boiler feed pump, inter-unit feedwater tie, emergency boiler cooling system/emergency water system, heat removal via the shutdown cooling system or maintenance cooling system or emergency cooling injection system) depending upon the needs of the accident. The design of these multiple systems and their support systems ensures that the heat removal function is available in operational states, DBAs and BDBAs.

As part of Fukushima enhancements and station improvements plans Bruce Power is making short-term provisions and longer-term provisions to provide make-up water to critical systems. Bruce Power has completed all short term modifications to allow emergency water to be added to the steam generators and IFBs using Emergency Mitigating Equipment (EME) pumps. Longer term provisions involve complementary design features which allow emergency makeup water to be added to the Bruce A and Bruce B Primary Heat Transport System and Moderator System and Shield Tank. The water is provided by portable EME pumps which are stored in a building adjacent to the site and at a higher elevation. Design Requirements have been established and the locations of the connection points for quick connect installation have been identified. A Preliminary Design Plan has been prepared and walkdowns have taken place. Installation of the connections will be linked to outage schedules. Installation in all units will be executed during unit outages starting in Q3-2017 [NK29-CORR-00531-14199].

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to S-142 Ultimate heat sink provisions

There were no strengths identified in SFRs from the standpoint of ultimate heat sink provisions.

As described in the review results above, GIO-002 Implement Design Changes to Improve Severe Accident Response will further improve ultimate heat sink provisions in extreme conditions in such events as earthquakes or severe accidents.

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

D-150 Design management

Principle: The assignment and subdivision of responsibility for safety are kept well defined throughout the design phase of a nuclear power plant project, and during any subsequent modifications.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 16 for DID Levels 1-4, as described below.

Levels 1 – 4

BP-PROG-10.01 Plant Design Basis Management ensures that the plant design meets safety, reliability, and regulatory requirements including pressure boundary quality assurance requirements as described in BP-PROG-00.04 Pressure Boundary Quality Assurance Program. Additionally, this program sets out roles and responsibilities, as well as requirements for engineering analysis and documentation such that the adequacy of the design can be demonstrated.

The role of Design Authority is described in Section 4.3 of BP-PROG-10.01. The Design Authority Procedure, as documented in DIV-ENG-00009 outlines the processes by which the Chief Engineer and Senior Vice President, Engineering executes the role of Design Authority.

The Design Authority Procedure is owned by the Chief Engineer and Senior Vice President, Engineering. The Chief Engineer and SVP Engineering as the Design Authority for the site ensures a strong nuclear safety culture consistent with Guideline WANO GL 2006-02 “Principles for a Strong Nuclear Safety Culture”.

BP-PROC-00335 Design Management Procedure specifies the design activities and outputs that define and manage the Plant Design Basis such that the nuclear operating stations can operate safely and reliably for the duration of their design life. Design Management relies upon the implementing procedures of BP-PROC-00363 Nuclear Safety Assessment to ensure nuclear safety requirements are incorporated into the design. This procedure interfaces with the implementing procedures of BP-PROG-10.02 Engineering Change Control, to ensure the correct tools are used during design changes and modifications. This procedure interfaces with the implementing procedures of BP-PROG-10.03, Configuration Management, to ensure margins are managed. The Design Management procedure is owned by the Department Manager, Component Design and Design Programs.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-150 Design management

SF-03-S1	The quality of the programmatic documents (i.e., programs and procedures) for the equipment qualification process is very good, with interfaces with other station procedures well identified, recent revisions and updating for most procedures, and incorporation of issues identified in audits and self-assessments.
SF-03-S2	The IAEA OSART review of Bruce B completed in 2015 reviewed all aspects of the environmental qualification program and recognized its overall implementation as “good performance”. Therefore, the management of the EQ program is considered to be a strength in this report.
SF-08-S4	Bruce Power’s leading role in the modification of the 37-element fuel design (37M) ensured the requirements were understood and fully incorporated, thus ensuring integration of the design and manufacturing aspects from multiple vendors who supported the project. This strength was important in ensuring the safety improvement was completed on schedule, implemented to Operation’s satisfaction and as committed to the CNSC.

There is 1 GIO included in the IIP that will further improve design management.

GIO No.	GIO TITLE
GIO-081	Human Factors in Design of Nuclear Power Plants

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

D-154 Proven technology
Principle: Technologies incorporated into design have been proven by experience and testing. Significant new design features or new reactor types are introduced only after thorough research and prototype testing at the component, system or plant level, as appropriate.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 17 for DID Levels 1-4, as described below.

Levels 1 – 4

The SSCs important to safety have been in place at Bruce A and Bruce B for over 30 years are of proven design as evidenced by their safe and reliable performance. The Bruce A and Bruce B designs were based upon experience gained from earlier plants (NPD, Douglas Point, Pickering A). The systems, structures and components have been designed, fabricated and operated within the requirements of the applicable engineering codes and standards. The design, manufacture and construction have taken advantage of relevant industry standards and best engineering practices.

The development of strategies and programs to address in-service testing, maintenance, repair, inspection and monitoring is a necessary aspect of the plant design phase. The strategies and programs to be implemented for these in-service activities are developed so as to ensure that plant SSCs remain capable and available to perform their safety functions. The design incorporates provisions recognizing the need for in-service testing, maintenance, repair, inspection and monitoring, as well as to permit the repair, replacement and modification of those SSCs likely to require such actions, due to anticipated operating conditions. In addition, activities which need to be carried out during the construction and commissioning phases are identified, in order to provide a meaningful baseline data of the plant, at the outset of its operating life.

All current and future design changes are implemented in accordance with BP-PROG-10.01 Plant Design Basis Management, which governs BP-PROC-00335 Design Management, the latter of which interfaces with the implementing procedures of BP-PROG-10.02, Engineering Change Control. Design changes over the years have been based upon design improvements (e.g., in-core detector assemblies) that have been tested and proven prior to implementation. All future design changes will be in accordance with BP-PROG-10.01, Plant Design Basis Management, which governs BP-PROC-00335, Design Management, the latter of which interfaces with the implementing procedures of BP-PROG-10.02, Engineering Change Control. As described in BP-PROC-00335 Design Management, all modifications to plant systems, structures and components, including temporary modifications and complex tools with a significant impact on nuclear safety are subject to change control. Change control is also applied to changes or revisions that only involve design documentation, including instances where a design document is discovered to not align with the field configuration. The change control of engineering documentation is implemented through BP-PROG-10.02, Engineering Change Control and BP-PROG-10.03, Configuration Management. As an illustration of this process, early design modifications to 37 element fuel bundles and self-powered in-core detectors have undergone comprehensive testing at Chalk River Laboratories. During the early years of operation, both were examined extensively to demonstrate that they met their objectives.

In addition, Bruce Power makes use of the OPEX associated with the current designs to ensure that design features of the replacement components are improved to address performance issues associated with in-service degradation and ageing. For example, Bruce Units 1&2 feeder piping portions susceptible to Flow Assisted Corrosion (FAC) were replaced with higher chromium content material. Steam Generator (SG) Tube bundle design was improved with better support design and tubing resistant to SSC based on OPEX.

Bruce Power actively participates in the research and development activities in CANDU technology to improve plant operation, equipment performance and reliability and analytical capabilities and scientific codes used in engineering and safety analysis.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-154 Proven technology

SF-08-S4	Bruce Power's leading role in the modification of the 37-element fuel design (37M) ensured the requirements were understood and fully incorporated, thus ensuring integration of the design and manufacturing aspects from multiple vendors who supported the project. This strength was important in ensuring the safety improvement was completed on schedule, implemented to Operation's satisfaction and as committed to the CNSC.

There are no additional planned initiatives included in the IIP that will further improve proven technology.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

D-158 General basis for design

Principle: A nuclear power plant is designed to cope with a set of events including normal conditions, anticipated operational occurrences, extreme external events and accident conditions. For this purpose, conservative rules and criteria incorporating safety margins are used to establish design requirements. Comprehensive analyses are carried out to evaluate the safety performance or capability of the various components and systems in the plant.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 18 for DID Levels 1-4, as described below.

Levels 1 – 4

The original Bruce A and Bruce B designs were based on a set of regulations and rules that were in effect during the 1960s which were built on the lessons learned from the NRX accident, as well as Nuclear Power Demonstration CANDU (NPD2) and Douglas Point. Based on the OPEX and regulatory guidance at the time, conservative rules and criteria incorporating safety margins were used to establish design requirements. The plant is designed to cope with a set of events including normal conditions, anticipated operational occurrences, extreme external events, such as low probability seismic events, tornadoes, floods etc., and accident conditions.

The Bruce A and Bruce B Safety Reports contain a comprehensive set of analyses that have been carried out to evaluate the safety performance or capability of the various components and systems in the plant. These analyses intentionally involve conservative assumptions, which increase the severity of the predicted consequences and hence result in large margins of safety. For example for accidents involving pipe breaks, it is assumed that:

- Most key safety parameters are simultaneously at the limit of their operating envelope values (for example, fuel bundle and channel powers are both assumed to be at their respective licensing limits).
- Only the safety systems mitigate the accident. Any process action that would intervene to reduce the consequences is not credited in the analysis. (Process controls that are actively functioning at the time of the initiating event are assumed to continue to control).
- Only the backup trip is credited. Further, the two most effective shutoff rods are assumed unavailable or, for the case of the second shutdown system, certain injection nozzles are unavailable.
- Weather conditions are assumed that maximize the population dose: atmospheric inversion with a light wind blowing in the direction of highest population density. (Such an inversion actually occurs less than 10% of the time).
- Dose is calculated for the most susceptible individual (i.e., a six month old baby), at the exclusion boundary.

The deterministic safety analysis is documented in Part 3 of the Safety Report, which has been updated periodically. The Bruce A and Bruce B PRA includes Level 1 and Level 2 analyses. The Bruce A and Bruce B PRA models have been updated to reflect the plant as built and operated. The hazard assessments establish a list of relevant internal and external hazards that may affect plant safety. Bruce Power undertook, as part of its disposition of Fukushima Action Items, a re-evaluation of the site-specific magnitudes of each external event to which the plant might be susceptible, using modern calculations and methods; and an evaluation as to whether the current site-specific design protection for each external event so assessed is sufficient.

Bruce Power is addressing the need for additional complementary design features through evaluations and potential design improvements as part of Fukushima Action Items. The design features which are introduced to cope with beyond design basis accidents include design or procedural considerations, or both, and are based on a combination of phenomenological models, engineering judgments, and probabilistic methods. As part of the Bruce Power Station Improvement Plans – Fukushima enhancements, projects are underway to enhance the existing understanding of severe accident phenomena and SAMG capabilities. The scope of this work also involves improvement to understanding of severe accident phenomena and containment integrity.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-158 General basis for design

SF-05-S1	Bruce Power has established an integrated strategy to improve the deterministic safety analysis contained in the Safety Reports as part of its objective to reach compliance with CNSC REGDOC-2.4.1 to the maximum practicable extent over a defined transition period. Bruce Power DSA procedures have been revised in consideration of CNSC REGDOC-2.4.1 requirements and the industry Principles and Guidelines for DSA. Industry guidelines for Limit of the Operating Envelope (LOE)/Realistic Operating Envelope (ROE) and Best Estimate Analysis and Uncertainty (BEAU) methodologies are established. Moreover, “Derived Acceptance Criteria for Deterministic Safety Analysis” is issued as COG 13-9035. Bruce Power is leading or actively participating in all SRI activities.
SF-06-S1	Bruce Power has developed and implemented a process of continuous maintenance of the PRA model to ensure that the model is representative of the actual plant configuration and operation and testing at the station. This exceeds the requirement of CNSC REGDOC-2.4.2 (Clause 4.4) that the PRA models be updated every five years.
SF-08-S4	Bruce Power’s leading role in the modification of the 37-element fuel design (37M) ensured the requirements were understood and fully incorporated, thus ensuring integration of the design and manufacturing aspects from multiple vendors who supported the project. This strength was important in ensuring the safety improvement was completed on schedule, implemented to Operation’s satisfaction and as committed to the CNSC.

There are 13 GIOs that will further improve general basis for design as they are related to initiatives to align the Bruce A and Bruce B designs with modern regulatory documents, codes and standards.

GIO No.	GIO TITLE
GIO-001	Improve documented design basis
GIO-003	Assess pipe whip and jet impingement
GIO-005	Assess cyclic loads of pressure retaining components designed per ASME III or VIII
GIO-009	Update safety analysis to align with REGDOC-2.4.1
GIO-024	Enhanced Periodic Safety Review to Support Asset Management
GIO-083	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
GIO-089	Whole-Site Probabilistic Risk Assessment
GIO-097	Bruce A Legacy Registration- Implementation Projects

GIO No.	GIO TITLE
GIO-098	Bruce B Legacy Registration- Implementation Projects
GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	M/34720 Relief Valves For Overpressure Protection
GIO-102	I/63472 Remote Relief Valve Position Indication
GIO-103	Implementation of Asset Management Activities

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Thirteen GIOs were identified that will further improve the provisions for this safety principle.

D-186 Inspectability of safety equipment

Principle: Safety related components, systems and structures are designed and constructed so that they can be inspected throughout their operating lifetimes to verify their continued acceptability for service with an adequate safety margin.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 25 for DID Levels 1-4, as described below.

Levels 1 – 4

The Bruce A and Bruce B designs incorporate provisions for safety related SSCs so that they can be inspected and tested throughout their operating lifetimes to verify their continued acceptability for service.

Each process and nuclear measurement loop that is essential for the operation of a special safety system is redundantly designed, usually triplicated, such that a single loop component or power supply failure will not incapacitate or spuriously invoke operation of the special safety system. This triplication and redundancy also allows each channel to be tested, inspected or repaired as necessary without tripping the system.

Bruce A and Bruce B have extensive testing programs to demonstrate that the special safety systems meet their ongoing reliability requirements. Section 03.5 of the Bruce Operating Policies & Principles (OP&Ps) specifies that the testing program is required on any system which is not normally operating but is required to function, in the event of a system failure, to control reactor power, cool the fuel, or contain radioactivity. The inspection and testing programs for these systems are consistent with reliability objectives established in system design.

Design provisions are implemented to minimize the radiation doses to workers, as well as access to components and systems that require periodic inspections per Section 6.1 Fitness for

Service Program of the PROL and the accompanying Licence Condition Handbook (LCH) [5] [6] in accordance with CSA standards N285.4, N285.5 and N287.7.

Much of the equipment, both safety and process, was placed outside containment to allow on-power maintenance, inspection and testing to the extent possible. All safety system equipment that requires testing or maintenance is accessible on-power from outside containment (e.g., SDS1 and SDS2 instrumentation, poison tank sampling, shutoff rod drives, etc.). In general, for systems or structures that cannot be tested, inspection or monitoring programs are in place. For example, corrosion in systems is not measured directly, but is done through chemical sampling, metallurgical examination of the irradiated material samples removed from the reactor SSCs. Leak detection is also utilized to augment testing and inspection activities or situations where testing or inspection is not possible. For example, on-line monitoring of humidity in the annulus gas system augments the inspection activities associated with assuring structural integrity of pressure tubes and calandria tubes.

In addition, MCR outage provides an opportunity to gain access to SSCs that are difficult to inspect and maintain under normal planned outage plant configurations. There are a number of initiatives being planned to inspect, maintain and repair or replace such components as listed below.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-186 Inspectability of Safety Equipment

There were no strengths identified in SFRs from the standpoint of inspectability of safety equipment.

There are 5 GIOs included in the IIP that will facilitate improved inspectability of safety equipment as a result of the plant configuration during the MCR outages.

GIO No.	GIO TITLE
GIO-059	Calandria and Shield Tank Assembly Major Inspection
GIO-060	Preheater Inspections
GIO-062	PHT Pump Seal Bellows Replacement
GIO-065	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection
GIO-066	Pressurizer and Supports- Internal Inspection

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Five GIOs were identified that will further improve the provisions for this safety principle.

D-205 Startup, shutdown and low power operation
Principle: Components Structures and systems used during startup, low power and shutdown operations are designed to maintain or restore the reactivity control, decay heat removal and the integrity of the fission product barriers so as to prevent the release of radioactive material resulting from accidents initiated during those operations.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 37 for DID Levels 1-4, as described below.

Levels 1 – 4

Reactivity control can be maintained during startup, low power and shutdown of the reactor by RRS, SDS1 or SDS2. The safe reactor unit shutdown states under which the Special Safety Systems, the Reactor Regulating System and the Heat Transport Pump Trip System are no longer required are specified in the Bruce A and Bruce B OP&Ps. For example, if the regulating system is incapable of controlling bulk power, then the reactor shall promptly be placed in a guaranteed shutdown state. The guaranteed shutdown states are defined in the Bruce A and Bruce B OP&Ps. Since the primary function of the Moderator system during outages is to ensure that the reactor remains in a Guaranteed Shutdown State, the requirements for guaranteed shutdown state are specified in Operational Safety Requirements (OSRs) for Moderator System. The Moderator System OSRs present the safety limits applicable to Over-Poisoned Guaranteed Shutdown State and Drained Guaranteed Shutdown State. A minimum poison concentration, in addition to the minimum level requirement, is required to prevent re-criticality while the reactor is shut down and special safety systems and regulating systems may be out of service. In addition, Bruce A and Bruce B Level 1 PRAs cover both at-power and shutdown (outage) states to identify vulnerabilities during shutdown states.

Both SDS1 and SDS2 are capable of shutting the reactor down fast enough for all AOOs, DBAs such that specified limits are not exceeded. There is no recriticality following accidents. For SDS1, operator action can be credited after 15 minutes to augment the depth of shutdown. For SDS2, the shutdown depth is sufficient to keep the reactor shut down indefinitely for even the most reactive conditions of the core.

In addition to the normal heat removal via the Heat Transport System, Shutdown Cooling System and the Maintenance Cooling System are designed for removing decay heat from the reactor core. The reactor coolant system is a barrier to the release of radioactive fission products and is therefore designed to retain its integrity under normal and abnormal operating conditions.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-205 Startup, shutdown and low power operation

There were no strengths identified in SFRs from the standpoint of improve startup, shutdown and low power operation.

There are 9 GIOs included in the IIP that will further improve SSCs used during startup, shutdown and low power operation within the context of the SP D-205.

GIO No.	GIO TITLE
GIO-076	DCC Cables and WIBAs –Replacement
GIO-077	Moderator Heat Exchangers- Replacement
GIO-078	Maintenance Cooling Heat Exchanger- Replacement
GIO-086	PHT Valves-Refurbishment of 33120-MV23
GIO-090	SDS2 Enhancements
GIO-099	Install Correctly Sized Maintenance Cooling Relief Valves
GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	M/34720 Relief Valves For Overpressure Protection
GIO-102	I/63472 Remote Relief Valve Position Indication

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Nine GIOs were identified that will further improve the provisions for this safety principle.

D-227 Monitoring of plant safety status

Principle: Parameters to be monitored in the control room are selected, and their displays are arranged, to ensure that operators have clear and unambiguous indications of the status of plant conditions important for safety, especially for the purpose of identifying and diagnosing the automatic actuation and operation of a safety system or the degradation of defence in depth.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 47 and 48 for DID Levels 1&2 and 3&4, respectively, as described below.

Levels 1 & 2

Instrumentation and control is centered around a dual, digital computer system that is used on each unit for control, alarm annunciation, data display and data logging. Direct digital control is used for such functions as regulating reactor power and steam generator pressure.

All functions essential to the operation of the unit are incorporated in both computers. Other functions, not essential to unit operation, may be resident in one computer only.

Unit signals are continually monitored and alarm messages are provided with an audible warning when limits are exceeded. The alarm messages are presented on two video displays, which may be read from the operation desk in the control room and are logged on line printers. As well, an option has been installed to store alarm messages on the moving arm disc only instead of being printed. The messages may be printed out when required. Alarm summaries may be requested for the total unit or on a system basis. A sequence of events record is produced following all unit disturbances.

A computer-driven video display system is provided in the control room. Standard displays in the form of equipment status, numerical data, bar charts or trend plots may be obtained from station inputs to the computers. Special display formats are provided for particular plant systems to improve process data presentation. Historical information for a number of inputs is saved on the moving arm disc and may be displayed at any time. A hard copy facility is provided to enable the operator to record display information for later use.

Keyboards mounted in the control room panels and on the operator’s desk provide the means for operator communication with the computers.

Bruce Power's Human Factors program is integrated into the design change process. Any on-going system changes that necessitate changes in the Main Control Room are addressed through the Human Factors program described in Human Factors Engineering Program Plan, DPT-PDE-00013. Human-machine interfaces, human information needs and workload are addressed in the Human Factors Engineering Program Plan, DPT-PDE-00013, which is supported by various Design Guides associated with specific plant systems.

Bruce Power provisions for Levels 1 and 2 for this safety principle are effective and adequate.

Levels 3 & 4

A computer-driven video display system is provided in the control room. Standard displays in the form of equipment status, numerical data, bar charts or trend plots may be obtained from station inputs to the computers. Special display formats are provided for particular plant systems to improve process data presentation. Historical information for a number of inputs is saved on the moving arm disc and may be displayed at any time. A hard copy facility is provided to enable the operator to record display information for later use.

Keyboards mounted in the control room panels and on the operator’s desk provide the means for operator communication with the computers.

The instrumentation and control systems for parameters to be monitored in the control room are designed to a large variety of detailed requirements, depending on their function, importance and physical environment. However, all the systems are designed to the following general criteria:

- The maximum practical amount of automatic control is incorporated in the design to allow the station to be operated safely with a minimum staff and to leave operators free for higher level monitoring of overall unit status. The operator can readily intervene in the operation of the automatic control systems.
- Adequate, comprehensive information is designed to be readily available at all times to allow the operator to assess the status of the unit quickly and to intervene with manual actions if necessary.
- Equipment is designed for a minimum of regular maintenance. Any necessary maintenance operations are kept as simple and efficient as possible.

- The instrumentation and control systems are designed for a very high reliability and availability, both to maximize plant availability and for safety. This reliability is achieved through a combination of component selection and design, and through redundancy
- The control systems are designed to make the unit as tolerant as possible to expected and unexpected transients, in order to prevent unnecessary unit outages.
- Where possible, the control systems are designed to prevent or minimize damage to equipment.

Secondary Control Areas (SCAs) are provided for post-accident monitoring and to execute basic safety functions following any common mode incident that renders the main control room uninhabitable. Follow-up by Bruce Power to the Nuclear Industry's lessons learned to the Fukushima-Daiichi severe accident has resulted in number of changes and significant improvement to Bruce Power's emergency preparedness capability. These changes include improved assessments and assurance of instrumentation and equipment survivability and plant habitability following severe accidents; improvements in communications capability both within the site and with outside agencies and, development of emergency response simulation software to enhance training for greater understanding and situation awareness of event response.

Bruce Power provisions for Levels 3 and 4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-227 Monitoring of plant safety status.

SF-02-S1	The conditions of the U014 and U058 SSCs are now tracked in SHRs. Bruce Power continues to improve and streamline the SHR processes as part of ageing and asset management, integrating these improvements with their anticipated obsolescence, testing, inspection and maintenance programs.
SF-02-S2	Bruce Power's preventive maintenance implementation is a station priority. The station management team monitors implementation and leaders enforce accountability

There is 1 GIO included in the IIP that will further improve monitoring of plant safety status.

GIO No.	GIO TITLE
GIO-076	DCC Cables and WIBAs –Replacement

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

D-230 Preservation of control capability

Principle: The control room is designed to remain habitable under normal operating conditions, anticipated abnormal occurrences and accidents considered in the design. Independent monitoring and the essential capability for control needed to maintain ultimate cooling, shutdown and confinement are provided remote from the main control room for circumstances in which the main control room may be uninhabitable or damaged.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 49 for DID Levels 1-4, as described below.

Levels 1 – 4

The Main Control Room is designed to remain habitable during normal operation. It is protected from the effects of the steam environment following a steam or feedwater line break. It is adequately protected from the effects of radiation following all design basis accidents due to the addition of shielding around the Emergency Coolant Injection (ECI) recirculation lines. Fire protection of the Main Control Room is also enhanced.

At Bruce A, as a result of the Units 3 and 4 Seismic Margin Assessment, upgrades for the control room ceiling were implemented by installing restraining ties to attach each diffuser panel to the main support runners, and installing retaining tabs on each of the glass panels. The upgrades prevent the ceiling tiles and glass panels from falling. Thus, the Main Control Room is now considered seismically qualified to the extent necessary to ensure the success path to maintain control, cooling and contain functions during a seismic event. The original seismic qualification of the Bruce B followed the criteria of Seismic Qualification of Safety-related Systems, [NK29-DG-03650-002].

Bruce Power is performing assessments, in conjunction with COG, of instrument survivability and habitability of control facilities under severe accident conditions and identification of modifications required is underway. Control facilities include areas in the field where SAMG operator actions are required.

The Secondary Control Area (SCA) is provided for post-accident monitoring and to execute basic safety functions following any incident that renders the main control room uninhabitable due to fire, smoke, or excessive radiation fields. All safety functions that are initiated automatically in the MCR can also be manually initiated within the SCA.

At Bruce A, there is one SCA located in the Construction Retube Building (CRB) which covers Units 1 and 2 and one SCA located in Unit 3 which covers Units 0, 3 and 4. The SCAs are physically separated and isolated from the main control room. At Bruce B, there are Secondary Control Areas in each of the four reactor buildings. They are physically separated and isolated from the main control room. There is also a secondary control area in the Emergency Water and Power Supply Building (EWPSB common SCA) at Bruce B. Control devices located in any SCA override the equivalent main control room controls.

The SCA permits control and monitoring of:

- Station emergency water and power systems.

- SDS2, neutronic safety parameters (Log-N and Log-N rate).
- Process safety parameters.
- Station emergency coolant injection.
- Containment.

The operator can make emergency announcements from the main control room through the public address system. Communications in the SCAs are facilitated through an internal/external telephone system and UHF radios. There is a direct dialing telephone system for communication between all areas of the station. The system is interconnected with the external public telephone network. A separate telephone system in the control room provides communications in the event of a failure of the station telephone system. The system is interconnected with the station telephone system, the other station's control room telephone system and the external public telephone network. A satellite telephone system is installed to provide the unit operator with a communications link independent of the public switched telephone system, the system voice circuits, and Class IV power. The installed equipment uses the MSAT telephone system adopted by the Independent Market Operator and interconnected utilities.

SCAs are seismically and environmentally qualified to provide control and indications that enable the operators to ensure the reactor units are shut down and monitored; the reactor units are cooled down and monitored and common containment is maintained and monitored. Seismically qualified egress routes are provided from the Main Control Room to the SCAs.

Bruce Power Severe Accident Management Guidance Plant Habitability - Summary Report was included as Enclosure 3 of [NK21-CORR-00531-11801 / NK29-CORR-00531-12195]. This assessment followed the methodology developed by COG. The results concluded, in Section 9.1.1, that overall for single unit accidents were found to be well mitigated for both Bruce A and Bruce B with respect to dose conditions in key areas surrounding the plant. For multiunit accidents it was demonstrated that the U0 SCA for Bruce B remains habitable until approximately 48 hours and the Main Control Rooms remain habitable for approximately 14 hours following a four-unit severe accident were Emergency Moderator Makeup (EMM) is not credited until after core collapse occurs (Section 9.1.2).

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

<u>Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-230 Preservation of control capability</u>

There were no strengths identified in SFRs from the standpoint of preservation of control capability.

There are 2 GIOs included in the IIP that will further improve Preservation of control capability.

GIO No.	GIO TITLE
GIO-090	SDS2 Enhancements
GIO-095	45VDC Power Supplies-Replacement

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

M&C-246 Safety evaluation of design

Principle: Construction of a nuclear power plant is begun only after the operating organization and the regulatory organization have satisfied themselves by appropriate assessments that the main safety issues have been satisfactorily resolved and that the remainder are amenable to solution before operations are scheduled to begin.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 55 for DID Levels 1-4, as described below.

Levels 1 – 4

Bruce A and Bruce B construction was initiated and completed in accordance with the regulatory requirements in effect at the time. The construction phase of the plants were supported by submitting appropriate assessments on the all relevant safety issues to the regulatory authority having jurisdiction for their satisfactory resolution and ensuring that any remaining issues were amenable to solution before operations were scheduled to begin.

In addition, safety evaluation of the design activities is an on-going activity through an extensive number of programs and procedures as part of compliance with the PROL. Design activities are performed in accordance with a comprehensive set of codes and standards in accordance with the PROL and best practices. Current design of the plants as documented in the Safety Reports, which are updated periodically, include safety features of the original plants, as well as safety improvements implemented in response to emerging issues and OPEX since they were put in service.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to M&C-246 Safety Evaluation of Design

There were no strengths identified in SFRs from the standpoint of preservation of control capability.

There are 3 GIOs included in the IIP that will further improve Safety Evaluation of Design.

GIO No.	GIO TITLE
GIO-009	Update safety analysis to align with REGDOC-2.4.1
GIO-083	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2

GIO No.	GIO TITLE
GIO-089	Whole-Site Probabilistic Risk Assessment

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Three GIOs were identified that will further improve the provisions for this safety principle.

M&C-249 Achievement of quality
Principle: The plant manufacturers and constructors discharge their responsibilities for the provision of equipment and construction of high quality by using well proven and established techniques and procedures supported by quality assurance practices.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 56 for DID Levels 1-4, as described below.

Levels 1 – 4

At the time Bruce A and subsequently Bruce B was built, the general design philosophy was to provide high quality process systems, with special emphasis on pressure retaining components which was reinforced by the NRX accident.

In 1972, D. Hurst (then president of the Atomic Energy Control Board) wrote, “The first line of defence in ensuring the low probability of accidents is the requirement for thoroughness and extremely high quality in the design and construction of the plant. Where applicable, the best appropriate standards and codes are required. Quality assurance systems are necessary to control procurement, construction, and installation”. This is particularly important for the physical barriers surrounding the radioactive material in the fuel.

Design and quality assurance processes were put in place for design analysis, stress analysis, material control and traceability, fabrication, in-process inspection, installation and welding, control of weld quality, Non-Destructive Examination (NDE), and inspection. This evolved into one of the first manufacturing based, comprehensive QA standards in the world. This nuclear industry standard required that suppliers have a level of QA commensurate with the nature and application of the goods being supplied. These standards formed the basis of the Z299 series of Canadian National Standards initially issued in the 1970’s under the Nuclear Standards Steering Committee. They were found useful in other industries and were taken over by the Quality Standards Steering Committee. Later, the international community decided that these standards and similar ones in other countries should be harmonized. Using these as the basis, the International Standards Organization (ISO) produced ISO 9000, now used extensively in a wide range of industries around the world.

Bruce Power governance includes a set of programs and procedures that enable a managed process of creating procurement specifications and plant configuration control as required for materials, systems and components. These programs and procedures are in accordance with the applicable requirements of CSA-N286 Management system requirements for nuclear power plants. The Procurement Engineering function specifies clear and adequate quality and technical requirements and activities that interface with other programs within the Supply Chain in order to ensure purchased items perform their intended end use design function(s). In-process inspections are performed using Inspection and Test Plans for all installations and may commence as equipment and components are installed, or may be scheduled when a specific area or system subsection is completed. Installation activities are tracked and verified in accordance with the work package.

BP-PROG-05.01, Supply Chain Program defines the requirements and responsibilities associated with the Supply Chain processes to ensure compliance with CSA N286-05. Elements of the program include: Procurement of Items and Services; Contract Management; Warehouse Operations; and Quality Oversight. The Quality Oversight is implemented through BP-PROC-00854, Quality Oversight.

The process of employing external technical, maintenance and other specialist staff is controlled through BP-PROG-02.01, Worker Staffing and BP-PROC-00355, Hiring Process (Contractors). BP-PROG-02.01 defines requirements for hiring of Regular, Temporary and Contract Employees. This process covers both contracted staff working within a defined scope project under the direct supervision of Bruce Power, as well as those personnel working under the supervision of an external organization that has been contracted to deliver a service. Contractors who work on site under Bruce Power supervision are required to attend the same orientation and training as a regular hired employee of Bruce Power. The Contract Manager or delegate is responsible for meeting with the successful bidder and identifying Bruce Power requirements for contractors accessing the Bruce Power site. BP-PROC-00041, Contract Management outlines the process utilized during the selection process for contractors. Specific controls are defined for contractors whose scope of work includes activities relating to nuclear safety or pressure boundary work.

All records are managed according to Records Management procedure BP-PROC-00098 to ensure all records regardless of media are properly categorized. BP-PROC-00972, Records Retrieval and Secure Storage, defines the controls for storage of and access to Bruce Power records to ensure their integrity and protection against damage, deterioration or loss. Records are stored in a predetermined storage facility for the retention period specified for each record. The retention process for Bruce Power records follows the steps outlined in BP-PROC-00238 for Bruce Power Records. The control and tracking of records is performed through the PassPort system with the most current documentation readily available to all users.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to M&C-249 Achievement of quality

There were no strengths identified in SFRs from the standpoint of achievement of quality within the context of this SP.

There are 7 GIOs included in the IIP that will further improve achievement of quality within the context of this SP.

GIO No.	GIO TITLE
GIO-001	Improve documented design basis
GIO-097	Bruce A Legacy Registration- Implementation Projects
GIO-098	Bruce B Legacy Registration- Implementation Projects
GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	M/34720 Relief Valves For Overpressure Protection
GIO-102	I/63472 Remote Relief Valve Position Indication
GIO-103	Implementation of Asset Management Activities

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Seven GIOs were identified that will further improve the provisions for this safety principle.

C-255 Verification of design and construction

Principle: The commissioning programme is established and followed to demonstrate that the entire plant, especially items important to safety and radiation protection, has been constructed and functions according to the design intent, and to ensure that weaknesses are detected and corrected.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 57 an 58 for DID Levels 1-3 and 4, respectively, as described below.

Levels 1 – 3

The original reactor systems were designed by AECL while Ontario Hydro Design and Construction Branch designed the balance of plant. From the earliest stages of the design, operating staff was assigned to the design organizations to make sure that appropriate input was provided to ensure that operating needs were dealt with. The design organization provided appropriate System Design Manuals to the operations staff prior to start up. From these manuals the operating staff developed Commissioning Plans and Procedures, Operating Manuals and Maintenance Manuals, and undertook the full commissioning of the station. The system design manuals provided operational limits for the various system components and the safety analysis provided safety limits for incorporation into the OP&Ps and Impairment Manual.

Both Bruce A and Bruce B were commissioned in accordance with the regulatory requirements in place prior to start of power operation. All SSCs were tested pre-operationally and commissioned individually or at the system level based on test plans and commissioning manuals in place. Commissioning activities included measurements to support evaluation of the accuracy of various safety analysis toolsets. For example, power rundown transient measurements during the Phase B commissioning experiments were used to demonstrate that the SDS2 power rundown rate is faster than the SDS1 power rundown rate. The systems Design Manuals require commissioning tests to be carried out to demonstrate that all parts and functions of the system meet their design requirements under normal conditions. In addition, tests also to be carried out as far as practical to determine that the system acts in a predicted and acceptable manner for faulted conditions (e.g., transient or temporary loss of power).

The Bruce Power Engineering Change Control Program BP-PROG-10.02, and Commissioning Modifications and Projects process, as documented in BP-PROC-00615, specifies how commissioning is to be carried out for Bruce Power Structures, Systems, Components and significant Tools. It includes requirements for commissioning planning, specification, execution, and reporting.

BP-PROC-00335, Design Management, requires applicable design inputs to be appropriately specified in a timely manner, documented and correctly translated into design output documents. These design inputs form the bases for design decisions, and their selection and modification is reviewed, verified and approved by the responsible design organization. The requirements for design verification and technical reviews are specified in Section 4.5 of the Design Management Procedure, BP-PROC-00335 as follows: Design verification ensures, through the process of reviewing, confirming, or substantiating design by one or more methods, that design meets specified design inputs, is technically adequate, and fulfils established design process requirements. Verification activities, including independence, qualification of staff, documentation of results, correction of deficiencies and specialized Technical Reviews are covered in DPT-PDE-00007, Design Verification. The Design Authority is responsible for undertaking the task of ensuring that all such interactions have been accounted for. The Nuclear Oversight Group, through their oversight role ensures that the process is being followed.

Bruce Power provisions for Levels 1-3 for this safety principle are effective and adequate.

Level 4

Both Bruce A and Bruce B were commissioned in accordance with the regulatory requirements in place prior to start of power operation. All SSCs were tested pre-operationally and commissioned individually or at the system level based on test plans and commissioning manuals in place. Commissioning activities included measurements to support evaluation of the accuracy of various safety analysis toolsets. For example, power rundown transient measurements during the Phase B commissioning experiments were used to demonstrate that the SDS2 power rundown rate is faster than the SDS1 power rundown rate. The systems Design Manuals require commissioning tests to be carried out to demonstrate that all parts and functions of the system meet their design requirements under normal conditions. In addition, tests are also required to be carried out as far as practicable to determine that the system acts in a predicted and acceptable manner for faulted conditions (e.g., transient or temporary loss of power).

The Bruce Power Engineering Change Control Program BP-PROG-10.02, and Commissioning Modifications and Projects process, as documented in BP-PROC-00615, specifies how commissioning is to be carried out for Bruce Power Structures, Systems, Components and significant Tools. It includes requirements for commissioning planning, specification, execution, and reporting.

The expectation is that commissioning will demonstrate that:

- Installed systems, equipment and components will perform in accordance with specifications and design intent before they are placed into service.
- Systems, equipment and components, which were altered to facilitate a change, are returned to their original configuration.
- Commissioning results are properly documented.
- Systems, equipment and components are ready for turnover.

The Engineering Change Control Program is implemented by the following procedures:

- BP-PROC-00539, Design Change Package
- BP-PROC-00542, Configuration Information Change
- BP-PROC-00615, Commissioning Modifications and Projects
- BP-PROC-00743, Site Services Engineering Change Control
- BP-PROC-00877, Modification Installation Quality Assurance

Bruce Power provisions for Level 4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to C-255 Verification of design and construction

There were no strengths identified in SFRs from the standpoint of verification of design and construction within the context of this SP.

There are 7 GIOs included in the IIP that will further improve verification of design and construction.

GIO No.	GIO TITLE
GIO-001	Improve documented design basis
GIO-097	Bruce A Legacy Registration- Implementation Projects
GIO-098	Bruce B Legacy Registration- Implementation Projects
GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	M/34720 Relief Valves For Overpressure Protection
GIO-102	I/63472 Remote Relief Valve Position Indication
GIO-103	Implementation of Asset Management Activities

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Seven GIOs were identified that will further improve the provisions for this safety principle.

C-258 Validation of operating and functional test procedures
Principle: Procedures for normal plant and systems operation and for functional tests to be performed during the operating phase are validated as part of the commissioning programme.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 59 for DID Levels 1-4, as described below.

Levels 1 – 4

The original reactor systems were designed by AECL while Ontario Hydro Design and Construction Branch designed the balance of plant. From the earliest stages of the design, operating staff was assigned to the design organizations to make sure that appropriate input was provided to ensure that operating needs were dealt with. The design organization provided appropriate System Design Manuals to the operations staff prior to start up. From these manuals the operating staff developed Commissioning Plans and Procedures, Operating Manuals and Maintenance Manuals, and undertook the full commissioning of the station. The system design manuals provided operational limits for the various system components and the safety analysis provided safety limits for incorporation into the OP&Ps and Impairment Manual.

Both Bruce A and Bruce B were commissioned in accordance with the regulatory requirements in place prior to start of power operation. All SSCs were tested pre-operationally and commissioned individually or at the system level based on test plans and commissioning manuals in place. Commissioning activities ensured that plant operating procedures and functional tests to be performed during the operating phase were validated as part of the commissioning activities.

When plant modifications are implemented, any procedural changes for associated SSCs are implemented through the ECC process and in accordance with BP-OPP-00001-Operating Policies and Principles for Bruce B and BP-OPP-00002-Operating Policies and Principles for Bruce A.

To support learning and qualification, Bruce Power has a variety of training facilities. The training facilities are designed to encourage dynamic learning and as a result incorporate numerous simulators and mock-ups, which include full scope simulators (two for Bruce A and one for Bruce B), Fuel Handling simulator shared by both stations, crane simulator, classroom simulators, live fire mock-ups, rescue training mock-ups, and maintenance shops for electrical, instrumentation and control, electronics, and maintenance training. The full scope main control room simulators are used for initial certification training of Bruce Power station staff, examination of staff, and continuing training of certified staff.

Bruce Power's Simulator Validation document, SEC-SIMM-00001, establishes the validation procedure for the full scope control room simulator. The validation procedure is used to confirm that the full scope simulators are capable of providing the correct observable simulated control room responses during the training and testing exercises. The Design Change Package process, BP-PROC-00539, ensures that changes to the plant are reflected in the main control room simulator. The Simulator Change Control, SEC-SIMM-00002, is used for documenting changes to the simulator. These procedures provide instructions for development, review, verification, approval, installation, commissioning, and closeout of any modification to the simulator.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to C-258 Validation of operating and functional test procedures

There were no strengths identified in SFRs from the standpoint of validation of operational and test procedures within the context of this SP.

There are no additional planned initiatives included in the IIP that will further improve validation of operational and test procedures within the context of this SP.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

C-260 Collection of baseline data

Principle: During commissioning tests, detailed diagnostic data are collected on components having special safety significance and the initial operating parameters of the systems are recorded.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 60 for DID Levels 1-4, as described below.

Levels 1 – 4

During construction and commissioning of Bruce A and Bruce B baseline data was collected in accordance with the inspection, testing and commissioning requirements to validate performance expectations, analytical tools and as a basis for comparison of plant critical parameters for future operation to ensure that plant is operated within its design basis. These activities involved collection of baseline data on special safety system components and overall system performance, as well as baseline NDE data on pressure boundary components and their supports. Design provisions are implemented to minimize the radiation doses to workers and to provide access to components and systems that require periodic inspections per N285.4, N285.5 and N287.7 which also covers gathering baseline data through inaugural inspections before the components are put in service.

Bruce Power also has programs and procedures in place for commissioning of modifications and projects to ensure design requirements and objectives are measured or proven. Allowable deviations, design limitation and assumptions are validated by demonstrating that the equipment or system has been installed as designed and performs as specified.

When plant modifications are implemented, any commissioning tests of SSCs are implemented through the Engineering Change Control (ECC) process and in accordance with BP-OPP-00001-Operating Policies and Principles for Bruce B and BP-OPP-00002-Operating Policies and Principles for Bruce A.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to C-260 Collection of baseline data

There were no strengths identified in SFRs from the standpoint of collection of baseline data within the context of this SP.

There are no additional planned initiatives included in the IIP that will further improve collection of baseline data within the context of this SP.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

C-262 Pre-operational adjustment of plant

Principle: During the commissioning programme, the as-built operating characteristics of safety and process systems are determined and documented. Operating points are adjusted to conform to design values and to safety analyses. Training procedures and limiting conditions for operation are modified to reflect accurately the operating characteristics of the systems as built.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 61 for DID Levels 1-4, as described below.

Levels 1 – 4

During construction and commissioning of Bruce A and Bruce B baseline data was collected in accordance with the inspection, testing and commissioning requirements to establish as-built operating characteristics of safety and process systems. These activities involved collection of baseline data on special safety system components and overall system performance, as well as baseline NDE data on pressure boundary components and their supports. These data were used to confirm that the future plant operation will be within the operating limits and conditions defined in design basis and supporting safety analysis. These data were also used as the basis for operating and training procedures.

Any adjustments to operating parameters and Operating Limits and Conditions (OLCs) are implemented through the ECC process and in accordance with BP-OPP-00001-Operating Policies and Principles for Bruce B and BP-OPP-00002-Operating Policies and Principles for Bruce A. BP-PROG-10.02 Engineering Change Control states that the Commissioning Modifications and Projects process, as documented in BP-PROC-00615, specifies how commissioning is to be carried out for Bruce Power Structures, Systems, Components and significant Tools. It includes requirements for commissioning planning, specification, execution, and reporting. The Design Change Package process, BP-PROC-00539, ensures that changes to the plant are reflected in the MCR simulator. The Simulator Change Control, SEC-SIMM-00002, is used for documenting changes to the simulator. These procedures provide instructions for development, review, verification, approval, installation, commissioning, and closeout of any modification to the simulator. The full scope main control room simulators are used for initial certification training of Bruce Power station staff, examination of staff, and continuing training of certified staff.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to C-262 Pre-operational adjustment of the plant

There were no strengths identified in SFRs from the standpoint of pre-operational adjustment of the plant within the context of this SP.

There are no additional planned initiatives included in the IIP that will further improve pre-operational adjustment of the plant within the context of this SP.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

O-269 Safety review procedures

Principle: Safety review procedures are maintained by the operating organization to provide a continuing surveillance and audit of plant operational safety and to support the plant manager in the overall safety responsibilities.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 63 for DID Levels 1-4, as described below.

Levels 1 – 4

The overall objective of BP-PROG-12.01 Conduct of Plant Operations is to safely and reliably operate the station systems within the design basis for which the plants are licensed. Surveillance of operational activities is performed in accordance with their safety significance by the operating staff and the requisite oversight is provided by responsible managers and supervisors in accordance with the implementing procedures of BP-PROG-12.01 Conduct of Plant Operations. For example, BP-PROC-00301 Reactivity Management states the following:

> "The purpose of this procedure is to establish principles and implement oversight of the activities that affect reactivity management at the stations. The goal of such oversight is to confirm that the reactor is always operated within the safe operating limits such that spatial or bulk loss of regulation will not occur. Compliance with these principles will also improve the overall reliability of the reactor units."

Self-evaluation activities include the completion of periodic State of the Functional Area (SOFA) Assessments, together with Focus Area Self Assessments (FASA), ad-hoc periodic reviews of trends, and oversight activities defined in implementing procedures. The SOFA Assessment process enables a standardized assessment across Functional Areas of the health of each Functional Area against Bruce Power's implementation of the Management System. BP-PROC-00137, Focus Area Self-Assessment, provides guidance in identifying and documenting lessons learned from internal sources to continuously improve performance by identifying weaknesses, strengths, threats and opportunities to make improvements to Processes/Procedures, Training, or System/Equipment Design. BP-PROC-00147, Benchmarking and Conference Activities, provides requirements for identifying and documenting lessons learned from external sources to continuously improve performance by making improvements to Processes/Procedures, Training, or System/Equipment Design.

BP-PROC-00059, Event Response and Reporting, defines the process for preliminary response and reporting to internal contacts and external agencies, to ensure compliance with both Bruce Power and Regulatory requirements for reporting OPEX.

BP-PROG-15.01, Nuclear Oversight Management, provides for the fundamental business need, constituent elements, functional requirements, implementing approaches, and key responsibilities associated with Nuclear Oversight Management in support of the plant managers in their overall safety responsibilities. It identifies the processes required to independently oversee the functioning of Bruce Power's Management System. This program contributes to the development and growth of Nuclear Safety Culture by communicating the Nuclear Safety message, setting the example for nuclear safety, and demonstrating this commitment through words and actions.

The Nuclear Oversight Management Program also serves to meet the embedded Power Reactor Operating Licence requirements for oversight of Pressure Boundaries and Environmental Protection.

The processing of internal or external events is administered using the Station Condition Record (SCR) process and the Corrective Action Program. BP-PROC-00060, Station Condition Record Process, is used by staff, including contractors, to document adverse conditions, investigation results and corrective actions related to people, plant, environment and process. Bruce Power processes related to Corrective Action are governed by the Corrective Action Program BP-PROG-01.07 and related implementing procedures. A Corrective Action Review Board (CARB), composed of senior management, performs a review of all significant events at Bruce Power.

Furthermore, the Bruce Power Board of Directors has established a Nuclear Safety Review Board, which has the responsibility for considering and advising the Board on the extent to which Bruce Power affairs are being conducted in a manner that promotes reactor, radiological, industrial and environmental safety and for continuing to emphasize the long-term effort required to improve safety culture permanently, including changing management behaviours and demonstrating leadership.

In addition, Bruce Power initiates and addresses the results of various detailed, confidential and privileged industry reviews conducted by organizations, such as WANO, INPO, and the IAEA (Operational Safety Review Team) reviews.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-269 Safety review procedures

SF-08-S1	An observed strength involves the commitments to improvements that are systematically being undertaken, based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently. (Same strength observed as in SF-10-S2 and SF-11-S2)
SF-10-S1	The existence of a comprehensive suite of programs and procedures that ensure the organization and administration will be controlled and well-documented in the future. Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes.

There are no additional planned initiatives included in the IIP that will further improve safety review procedures.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

O-292 Radiation protection procedures
Principle: The radiation protection staff of the operating organization establish written procedures for the control, guidance and protection of personnel, carry out routine monitoring of in-plant radiological conditions, monitor the exposure of plant personnel to radiation, and also monitor releases of radioactive effluents.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 69 for DID Levels 1-4, as described below.

Levels 1 – 4

Radiation Protection Procedures for Bruce A and Bruce B comply with the relevant provisions in the PROL and accompanying LCH [5] [6] including those in CSA N286-05 relating to radiation protection.

As defined in BP-MSM-1 Bruce Power Management System Manual, radiation protection (safety) is one of the four pillars of nuclear safety which supports a healthy nuclear safety culture.

BP-PROG-12.05 Radiation Protection Program defines the implementing approaches and key responsibilities associated with implementing the Radiation Protection Management Policy. This is achieved by establishing and implementing standards and processes for the conduct of licensed activities defined in Appendix A of the program document. A suite of program implementing procedures is in place to ensure public and occupational exposures to ionizing radiation are controlled so that:

- Individual doses are kept below regulatory dose limits;
- Unplanned exposures are avoided; and
- Individual and collective doses are maintained at levels As Low as Reasonably Achievable (ALARA), social and economic factors being taken into account.

The training requirements for workers to perform radiological work, requirements for Nuclear Energy Workers, and radiation protection qualification requirements for individuals to access and work at Bruce Power facilities are defined in BP-PROG-12.05, Radiation Protection Program. BP-PROG-12.05 also describes the procedures and processes in place to ensure radiological incidents are responded to promptly, and investigated to ensure the safety of all workers and the public.

Plant design features in support of the zoning and access control minimize the need for personnel to enter areas with high radiation fields. Extensive use is made of physical barriers, permanent and temporary signs, and other means to clearly warn and instruct personnel of any possible danger from radiation. In addition, operational procedures restrict access to the reactor building to qualified personnel and those escorted by qualified personnel. Access to areas that either have or could have high radiation fields is strictly controlled by the Access Control System. Access controlled areas have locks and keys controlled by the shift manager. All access to access controlled areas is allowed only on the basis of approved Work Authorizations, and a formal written and approved request for the issuance of an access control area key.

Bruce Power's Nuclear Emergency Response Plan, BP-PLAN-00001, describes the concepts, structures, roles and processes needed to implement and maintain Bruce Power's radiological emergency response capability. The NERP applies to all facilities within the Bruce Power Site and is developed to support response to design basis accidents that endanger the safety of personnel in the incident station, personnel on-site, members of the public and the environment, but also takes into account requirements to support a sustained response to Beyond Design Basis events. The NERP predominantly deals with releases of radioactive materials from fixed facilities; however, the infrastructures that are defined within this plan can be used to support the planning and response to all emergencies at the Bruce Power site.

Radiological releases to the environment and subsequent doses to the public are estimated by the Chemistry and Environment Departments and controlled by Plant Operations. The requirements for these controls are documented in BP-PROG-00.02 Environmental Safety Management, BP-PROG-12.01 Conduct of Plant Operations; and BP-PROG-12.02 Chemistry Management.

BP-PROC-00171, Radiological Emissions Monitoring: Limits, Action Levels, describes the Bruce Power expanded framework for control of radioactive emissions from Bruce A, Bruce B, and the Central Maintenance and Laundry Facility for the radiological protection of the public and the environment. This procedure defines derived release limits, action levels, internal investigation levels, and normal operating levels and describes associated processes used to assure emissions are managed to As Low As Reasonably Achievable (ALARA).

BP-PROC-00080 Effluent Monitoring Program sets the requirements for recordkeeping. Results of effluent monitoring are reported in quarterly operations reports to the CNSC in accordance with the requirements of PROL and the accompanying LCH [5] [6].

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

<u>Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-292 Radiation protection procedures</u>

SF-15-S1	Bruce Power has a mature and comprehensive radiation protection program that, by 2009, had begun to show the effects of aging and lack of maintenance. This contributed to the loss of RP controls observed during the 2009 Alpha Contamination Incident. Since that time, Bruce Power has made progress in addressing the deficiencies through RP improvement and excellence programs (see Section 4.2 of Bruce A SFR 15). Bruce Power recognized that significant change was required in all areas of RP at Bruce Power, and acted on this by developing extensive RP improvement initiatives and significantly reorganizing the RP Department at each of the Bruce Power facilities. Bruce Power has since improved and leads the way in the performance indicator for Collective Radiation Exposure (CRE) in North America. This industry-leading CRE performance has been identified as a strength in performance.

There are 3 GIOs that will further improve radiation protection procedures as they are related to initiatives to align Bruce A and Bruce B with WANO GL 2004-01 (2004) Guideline for Radiological Protection at Nuclear Power Stations.

GIO No.	GIO TITLE
GIO-082	Performance testing of nuclear air-cleaning systems
GIO-093	RP equipment and instrumentation maintenance and life cycle management
GIO-094	Effective use of the action tracking system in Radiation Protection

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Three GIOs were identified that will further improve the provisions for this safety principle.

O-299 Feedback of operating experience

Principle: Plant management institutes measures to ensure that events significant for safety are detected and evaluated in depth, and that any necessary corrective measures are taken promptly and information on them is disseminated. The plant management has access to operational experience relevant to plant safety from other nuclear power plants around the world.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 71 for DID Levels 1-4, as described below.

Levels 1 – 4

As illustrated in Figure 25 (see page 103) Bruce Power management has instituted programs and procedures to ensure that events significant for safety are detected and evaluated in depth and that any necessary corrective measures are taken promptly and information on them is disseminated through BP-PROG-01.07 Corrective Action and its implementing procedures.

BP-PROG-01.06 Operating Experience Program and its implementing procedures ensure access to operational experience relevant to plant safety from other nuclear power plants in Canada and around the world. The OPEX program covers both internal and external operating experience. The OPEX program and Corrective Action Program BP-PROG-01.07 are closely inter-connected and complementary. BP-PROC-00062, Processing External and Internal Operating Experience, provides detailed instructions on how to extract and process incoming and outgoing OPEX. The processing of internal or external events is administered using the Station Condition Record (SCR) process and the Corrective Action Program. BP-PROC-00518, Root Cause Investigation, is used to identify the root cause of an event and incidents so proper corrective action is initiated to prevent the future reoccurrence of similar events and incidents. BP-PROC-00519, Apparent Cause Evaluation (ACE), defines the process for performing an ACE and an Equipment Apparent Cause Evaluation (EACE). BP-PROC-00644, Common Cause Analysis, is used on adverse trends so corrective action can be taken to reduce the probability of the adverse trend continuing. BP-PROC-00412, Trending, Analyzing, and Reporting of SCRs, determines whether performance is improving, declining or stagnant; and corrective actions are initiated to address adverse performance before a break-through event occurs.

Feedback from relevant Research and Development is evaluated in terms of impact on plant design basis and safety analysis through BP-PROG-10.01 Plant Design Basis Management, as well as BP-PROG-01.07 Corrective Action for any follow-up.

Bruce Power actively participates in national (CANDU Owners Group) and international (INPO, EPRI, WANO) information sharing on Operating Experience and Research and Development. Bruce Power performs research in conjunction with the Ontario Ministry of the Environment and Climate Change, and attends workshops to acquire OPEX (e.g., Radiological Effluents and Environmental Workshop). Operating Experience and feedback from Research and Development are communicated to the CNSC through BP-PROG-06.03 CNSC Interface Management in accordance with the provisions of PROL [5].

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

<u>Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-299 Feedback of operating experience</u>

SF-08-S3	Bruce Power’s organization shares Safety Performance OPEX, Compliance Reporting and Corrective Action processes as commonly-maintained programs with Bruce B, and thus observations and lessons learned at Bruce B can be used at Bruce A. Additionally, there is an opportunity to share knowledge from Bruce B by transferring managers to Bruce A and vice-versa. Thus, strengths at each station and means to see how the other Station prevents and mitigates less desirable situations are shared to increase the corporate knowledge and experience. (Same strength observed as in SF-10-3).
SF-09-S1	The review demonstrates that Bruce Power’s OPEX Program and its implementation provides for adequate feedback of safety experience from nuclear power plants (both internal and external) and of the findings of research in support of continued safe and reliable operation. In addition, the review demonstrates that Bruce Power does not confine itself to utilizing OPEX from nuclear power plants only, but makes use of OPEX from any industrial process plants. Moreover, research activities are being pursued and results are used to enhance nuclear safety and equipment performance and reliability. This is regarded as a strength in Bruce Power’s OPEX Program.

There are no additional planned initiatives included in the IIP that will further improve feedback of operating experience.

<u>Summary</u>

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

O-305 Maintenance, testing and inspection

Principle: Safety related structures, components and systems are the subject of regular preventive and predictive maintenance, inspection, testing and servicing when needed, to ensure that they remain capable of meeting their design requirements throughout the lifetime of the plant. Such activities are carried out in accordance with written procedures supported by quality assurance measures.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 72 for DID Levels 1-4, as described below.

Levels 1 – 4

As shown in Figure 10 and Figure 11 (see pages 57 and 59) regular preventive and predictive maintenance, inspection, testing and servicing of SSCs important to safety and reliability are conducted in accordance with BP-PROG-11.01 Equipment Reliability, BP-PROG-11.04 Plant Maintenance and BP-PROG-00.04 Pressure Boundary Quality Assurance Program.

Under the Equipment Reliability Program, BP-PROG-11.01, life-cycle management integrates ageing management and economic planning to optimize the service life of SSCs and maintain an acceptable level of performance and safety over the life of the plant. The implementing procedures deal with scoping and identification of critical SSCs, continuing equipment reliability improvement, preventive maintenance implementation, performance monitoring, equipment reliability problem identification and resolution, long-term planning and life-cycle management.

BP-PROG-11.04, Plant Maintenance defines the performance needs, requirements, implementing approaches and responsibilities of the management of the plant maintenance process. It covers the maintenance of plant SSCs based on the approved maintenance strategies, schedules, procedures and practices in a cost effective manner that maximizes the availability and reliability of safety related and production sensitive equipment while maintaining the commitment to Nuclear Safety: Reactor, Radiation, Environmental and Industrial Safety. Predictive and preventive maintenance supports enhanced equipment reliability and improved operational safety performance. Maintenance strategies are continually refined using improved technologies, OPEX and plant reliability integration feedback. Work selection, prioritization and response are guided by risk informed decision making.

BP-PROG-00.04, the Pressure Boundary Quality Assurance Program describes the program to control the quality of pressure boundary activities at the facilities. It complies with the applicable rules and quality assurance requirements contained in CSA Standards: a) N285.0 and supporting codes for Class 1, 1C, 2, 2C, 3, 3C, 4 and 6 systems and components, and b) B51 and supporting codes for Class 6 and unclassified registered systems and components. Pressure boundary activities are performed in accordance with the Codes and Standards required by the PROL.

All of these programs are supported by a set of detailed implementing procedures. The programs are integrated with Plant Design Basis Management, Engineering Change Control and Configuration Management Programs to confirm and ensure that they remain capable of meeting their design requirements throughout the lifetime of the plant.

BP-PROG-14.02, Contractor Management provides guidance to personnel acting as Contract Managers/Officers and Supervisors for accomplishing effective oversight of contractors and supplemental personnel performing work for Bruce Power. The program defines the roles and responsibilities of the Contract Manager/Officer, which includes the following:

- Responsible for the site administration, coordination and overall performance of the contractor while working at the site, including but not limited to: quality, timeliness, safety and error-free performance; and
- Ensures the contractor's personnel are qualified and trained to perform the work assigned including any additional risk based training that may be required for specific tasks.

BP-PROG-12.05, Radiation Protection Program defines the requirements and implementing approaches of the Radiation Protection Management Policy as defined in the Management System Manual (BP-MSM-1, Appendix A). This Program also defines the requirements for compliance with Ontario Occupational Health and Safety Act (OHSA), X-Ray Safety and Radiation Emitting Devices (RED) Act requirements. The Radiation Protection Program is applicable to all Bruce Power facilities and all workers performing radiological work at Bruce Power, whether they are full-time or part time-staff, or contractors.

The requirements established in BP-PROG-02.02, Worker Learning and Qualification Program apply to Bruce Power personnel and training areas (with the exception of Nuclear Security). BP-PROG-02.02 ensures that personnel are provided with the competencies and qualifications necessary to satisfy the requirements of applicable legislation and other regulatory requirements commensurate with Bruce Power business needs. The Bruce Power training processes follow a Systematic Approach to Training to meet the requirements of B-HBK-09500-00003, Training Performance Objectives and Criteria. This document contains standards for training intended to promote excellence in support of operating the Bruce Power nuclear generating stations.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

<u>Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-305 Maintenance, testing and inspection</u>

SF-02-S1	The conditions of the U014 and U058 SSCs are now tracked in SHRs. Bruce Power continues to improve and streamline the SHR processes as part of ageing and asset management, integrating these improvements with their anticipated obsolescence, testing, inspection and maintenance programs.
SF-02-S2	Bruce Power's preventive maintenance implementation is a station priority. The station management team monitors implementation and leaders enforce accountability
SF-04-S1	Information from the Asset Management Program is proactively used to inform the business of the future needs related to ageing and to ensure the funding and priorities can be proactively established as required to ensure effective ageing management and plant safety.
SF-04-S2	Bruce B is an industry leader in the area of managing obsolescence of technology as evidenced by being awarded a WANO Strength and being the subject of a WANO Good Practice publication

There are 22 GIOs that will further improve maintenance, testing and inspection. There are 16 GIOs that will be implemented as part of the MCR outage to ensure that SSCs remain capable of meeting their design requirements throughout the lifetime of the plant.

GIO No.	GIO TITLE
GIO-025	Perform R&D in support of fuel channel life cycle management initiatives
GIO-034	Safety System Reliability
GIO-039	Equipment Reliability and Maintenance
GIO-056	Fuel Channel Replacement
GIO-057	Steam Generator Replacement
GIO-058	Feeder Replacement
GIO-059	Calandria and Shield Tank Assembly Major Inspection
GIO-060	Preheater Inspections
GIO-062	PHT Pump Seal Bellows Replacement
GIO-064	Control Distribution Frame (CDF) Terminal Replacement
GIO-065	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection
GIO-066	Pressurizer and Supports- Internal Inspection
GIO-070	Air Operated Valves-Replacement
GIO-071	Large Motors-Refurbishment/Replacement
GIO-076	DCC Cables and WIBAs –Replacement
GIO-077	Moderator Heat Exchangers- Replacement
GIO-078	Maintenance Cooling Heat Exchanger- Replacement
GIO-086	PHT Valves-Refurbishment of 33120-MV23
GIO-093	RP equipment and instrumentation maintenance and life cycle management
GIO-095	45VDC Power Supplies-Replacement
GIO-103	Implementation of Asset Management Activities
GIO-104	Ongoing Work on Bruce B Heat Transport Vibration Project

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Twenty-two GIOs were identified that will further improve the provisions for this safety principle.

O-312 Quality assurance in operation
Principle: An operational quality assurance programme is established by the operating organization to assist in ensuring satisfactory performance in all plant activities important to plant safety.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 73 for DID Levels 1-4, as described below.

Levels 1 – 4

The BP-MSM-1 Bruce Power Management System (BPMS) serves as Bruce Power's Quality Assurance Program and, as such, conforms with the requirements of CSA N286-05, Management System Requirements for Nuclear Power Plants specified in the PROL and the accompanying LCH [5] [6]. A matrix outlining the alignment of Bruce Power Programs to N286-05 is included in Appendix B of BP-PROG-01.02 Bruce Power Management System (BPMS) Management. The N286-05 Compliance Matrix is reviewed biennially as part of the oversight activities associated with BP-PROG-01.02 Bruce Power Management System (BPMS) Management.

The objective of BP-PROG-01.02 Bruce Power Management System (BPMS) Management is to establish the framework for the planning, implementation, maintenance, and continual improvement of business processes, activities, and human behaviors which contribute to the achievement of Bruce Power's objectives, and enables all business, legal, regulatory and other requirements to be defined and achieved.

BP-PROG-15.01, Nuclear Oversight Management program identifies the processes required to independently oversee the functioning of Bruce Power's Management System. This program contributes to the development and growth of Nuclear Safety Culture by communicating the Nuclear Safety message, setting the example for nuclear safety, and demonstrating this commitment through words and actions. The Program also serves to meet the embedded PROL requirements for oversight of Pressure Boundaries and Environmental Protection. These are accomplished by the Planning, Scheduling, Conducting, Reporting, and Overall Evaluation of Audits and Assessments.

BP-PROG-03.01, Document Management defines the fundamental business need, constituent elements, functional requirements, implementing approaches and key responsibilities associated with the management of Controlled Documents and Records. Controlled Documents are subject to formal procedural control of their preparation, review, validation, approval, issue and change control. Controlled Documents are reviewed for accuracy and approved by authorized personnel prior to release. All records are managed according to Records Management procedure BP-PROC-00098 to ensure all records regardless of media are properly categorized. The control and tracking of records is performed through the PassPort system. The preparation, issue and change of documents that specify quality requirements or prescribe activities affecting quality are controlled to assure that correct documents are being employed. Such documents, including changes thereto are reviewed for adequacy and approved for release by authorized personnel.

Bruce Power provisions for Levels 1-4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-312 Quality assurance in operation

SF-10-S1	The existence of a comprehensive suite of programs and procedures that ensure the organization and administration will be controlled and well-documented in the future. Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes.
SF-11-S1	The existence of a comprehensive suite of programs and procedures that ensure procedures will be controlled and well documented in the future. Additionally, Bruce Power demonstrates a strong commitment to continuous improvement by conducting regular self-assessments of their processes and revision of their procedures to meet best industry practice. This Safety Factor 11 review found that all aspects of the processes are satisfactory.

There are 2 GIOs that will further improve quality assurance in operation.

GIO No.	GIO TITLE
GIO-088	Improve Licencing Processes
GIO-094	Effective use of the action tracking system in Radiation Protection

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3, and 4, which are the levels applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

18.3.3. Safety Principles Related to Defence-in-Depth Levels 1, 2, 3

There are 4 Safety principles related to DID Levels 1, 2 and 3. Bruce A and Bruce B design and operation are aligned with all the safety principles as demonstrated below.

D-192	Protection against power transient accidents
D-195	Reactor core integrity
O-278	Training
O-284	Operational limits and conditions

D-192 Protection against power transient accidents
Principle: *The reactor is designed so that reactivity induced accidents are protected against, with a conservative margin of safety.*

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 27, 28, and 29 for DID Levels 1, 2, and 3, respectively, as described below.

Level 1

Design for prevention of accidents focuses on continuously and reliably matching heat removal capability with fuel power production. Primary heat production and heat removal control are achieved using dual computers for critical functions such as reactor power control and boiler pressure control. The system consists of two independent computer channels, each capable of complete unit control. Each channel contains a digital control computer with annunciation and command processing. The software and hardware operations are continuously monitored by a combination of internal self-checking software and hardware plus an external watchdog timer. Detection of a serious fault in the control computer results in the transfer of control to the standby computer.

The reactor regulating system is designed to maintain overall reactivity control during normal operation and following a range of AOOs by controlling the light water level in the liquid zone controllers. The system includes the input sensors, the Digital Control Computer (DCC) programs, the reactivity control elements in the reactor and the associated control and display devices and the Setback and Stepback functions. The automatic computer controlled regulating system maintains flux shape control in the core by adjusting the water level in the 14 light water filled individual zone control units. The reactor control system is designed to control both core flux and process parameters to predetermined levels under normal operating conditions. The flux shapes in the core can be measured by detectors in the regulating system (process system) and in both shutdown systems (special safety systems). Different types of detectors are used in the process and safety systems and they are totally independent of each other, thereby ensuring that common mode failure of all detectors is very unlikely.

The RRS OSR and the Fuel and Reactor Physics OSRs specify Safety Analysis Limits on RRS reactivity device configurations. These requirements are developed based on Safety Analysis Limits, which are derived from the safety analysis and supporting documents. The Safety Analysis Limits define the minimum hardware functional and performance requirements and the limiting process parameter values in the hardware subsystems, and are used to ensure that there is sufficient margin to the nominal automatic actuation setpoints to account for instrument loop uncertainty.

Plant Chemistry Management Program, BP-PROG-12.02, has the objective to establish the optimum conditions for system chemistry and to mitigate conditions that could lead to an adverse effect on nuclear safety, radiological safety, personnel safety, environmental safety or plant condition.

The OP&Ps are defined to clearly outline operating boundaries within which the station may be operated safely. Within these boundaries, detailed operating procedures are written for clearly defined operating requirements. Procedures are also written for abnormal or emergency conditions which may be accurately defined.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Level 2

Primary heat production and heat removal control are achieved using dual computers for critical functions such as reactor power control and boiler pressure control. The system consists of two independent computer channels, each capable of complete unit control. Each channel contains a digital control computer with annunciation and command processing. The software and hardware operations are continuously monitored by a combination of internal self-checking software and hardware plus an external watchdog timer. Detection of a serious fault in the control computer results in the transfer of control to the standby computer.

The reactor regulating system is designed to maintain overall reactivity control during normal operation and following a range of AOOs by controlling the light water level in the liquid zone controllers. The reactor regulating system includes the input sensors, the DCC programs, the reactivity control elements in the reactor and the associated control and display devices and the setback and stepback functions. The flux shapes in the core can be measured by detectors in the regulating system (process system) and in both shutdown systems (special safety systems). The reactor has both vertical and horizontal in-core flux detectors. Different types of detectors are used in the process and safety systems and they are totally independent of each other, thereby ensuring that common mode failure of all detectors is very unlikely. Under certain transient conditions, i.e., AOOs, if the reactivity range of the liquid zone controllers is exceeded, then further control via the regulating system is through the use of the control absorbers. The safety analyses have demonstrated that the fuel either remains cool or cooling is re-established in the event of a Loss of Coolant Accident (LOCA) such that the allowable release limits are met for all AOOs and DBAs. The safety analyses have shown that even for the largest LOCA the fuel damage is limited and no failure of pressure tubes is predicted. Thus, the reactor core remains intact. In the case of a single channel failure, i.e., pressure tube / calandria tube (PT/CT) rupture, the dynamic forces resulting during the blow down cause some damage to the internal structures but enough shutoff rods remain intact to meet all the relevant requirements.

Bruce Power provisions for Level 2 for this safety principle are effective and adequate.

Level 3

The safety analyses have demonstrated that the fuel either remains cool or cooling is re-established in the event of a LOCA such that the allowable release limits are met for all AOOs and DBAs. The safety analyses have shown that even for the largest LOCA the fuel damage is limited and no failure of pressure tubes is predicted. Thus, the reactor core remains intact. In the case of a single channel failure, i.e., pressure tube / calandria tube (PT/CT) rupture, the dynamic forces resulting during the blow down cause some damage to the internal structures but enough shutoff rods remain intact to meet all the relevant requirements.

In the event of control system failure or any other event that causes a mismatch beyond the capability of primary control devices, independent fast acting shutdown devices operate to rapidly reduce reactor power. Section 4.2.6 of Part 2 of the Bruce A and Bruce B Safety Reports describes the two fully capable, separate, independent and diverse shutdown systems. Each system has its own initiation sensors, detectors and logic to ensure functional and physical diversity.

As described in Section 4.2.6 of Part 2 of the Bruce A and Bruce B Safety Reports, Bruce A SDS1 has 30 and Bruce B SDS1 has 32 neutron absorbing rods that are referred to as shutoff rods. SDS1 uses a spring assisted gravity rod drop as its actuation means. The rods are held out by a power driven clutch system and upon activation of the trip parameter, the power to the clutches is removed and the rods are inserted. Thus, no active power mechanism is required to insert the rods in SDS1.

The SDS2 Liquid Injection Shutdown System consists of an arrangement of horizontal tubes (7 at Bruce A and 8 at Bruce B) with nozzles that are designed to inject heavy water poisoned with gadolinium nitrate into the moderator. SDS2 utilizes rapid injection of concentrated gadolinium nitrate solution into the bulk moderator through seven horizontally distributed nozzles. The poison injection is driven by stored energy in high-pressure gas tanks. The pressure is applied to the poison tanks only after the activation signal is received. The high-speed injection valves are air to close valves. This again means that stored energy is used for activation of SDS2. SDS2 employs an independent triplicated logic system, which senses the requirement for emergency shutdown and opens fast acting valves to inject the gadolinium poison into the moderator using high pressure helium.

As stated in Section 4.1 of Part 2 of the Safety Report, the design of the Bruce B reactors is essentially the same as that of Bruce A. The major changes that were incorporated into the Bruce B design are increased shutoff rod depth for SDS1, the addition of five horizontal flux detector units, the addition of one extra injection nozzle and injection tank for SDS2, and the adoption of adjuster units in place of booster units.

Both shutdown systems are capable of shutting the reactor down fast enough for all AOOs and DBAs such that specified limits are not exceeded. There is no recriticality following accidents. For SDS1, operator action can be credited after 15 minutes to augment the depth of shutdown. For SDS2, the shutdown depth is sufficient to keep the reactor shut down indefinitely for even the most reactive conditions of the core.

As part of the LLOCA Safety Margin Restoration Project a number of design changes that can provide improvement to LLOCA safety margins have been identified. These alternatives include improving the effectiveness of both shutdown systems (SDSs) by adding two neutronic trips in each SDS to sufficiently reduce the trip time credited in safety analysis. The two new trips in each SDS are intended to make use of the existing neutronic signals with one trip using signals from the in-core flux detectors and the other from the ex-core ion chambers.

Bruce Power provisions for Level 3 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-192 Protection against power transient accidents

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There are 3 GIOs that will further improve protection against power transient accidents.

GIO No.	GIO TITLE
GIO-026	BA & BB New Neutronic Trips
GIO-036	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
GIO-076	DCC Cables and WIBAs –Replacement

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, and 3, which are the levels applicable to this safety principle. Three GIOs were identified that will further improve the provisions for this safety principle.

D-195 Reactor core integrity

Principle: The core is designed to have mechanical stability. It is designed to tolerate an appropriate range of anticipated variations in operational parameters. The core design is such that the expected core distortion or movement during an accident within the design basis would not impair the effectiveness of the reactivity control or the safety shutdown systems or prevent cooling of the fuel.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 30, 31, and 32 for DID Levels 1, 2, and 3, respectively, as described below.

Level 1

Section 4 of Part 2 of the Safety Reports describes the mechanical and nuclear design of the reactor. Additional details are provided in the design manuals for different components of the reactor.

The loading condition for each component is determined from the worst possible combinations of loads and temperatures to meet the requirements in the applicable code classes of Section III and Section VIII of the ASME Code.

The allowable deflection limits are established by the ASME code such that the allowable stresses remain within elastic limits except where acceptance of some permanent strain is necessary to be compatible with the functional requirements.

The component and system test pressures are established in accordance with the rules for the appropriate component Class of Section III of the ASME Code.

Table 4.4 of Part 2 of the Bruce A and Bruce B Safety Reports list the operating conditions that were considered in the design. The stress analysis of all systems and major components in the Heat Transport (HT) system meets the requirements of Section III of the ASME Code. The types of stress analysis employed are tailored to the particular requirements for each system and component, and are identified in the stress reports produced for Class 1 systems and components. The faulted conditions considered in the pressure boundary analysis are identified in the stress reports produced for systems and components.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Level 2

Section 4 of Part 2 of the Safety Reports describes the mechanical and nuclear design of the reactor. Additional details are provided in the design manuals for different components of the reactor.

The loading condition for each component is determined from the worst possible combinations of loads and temperatures to meet the requirements in the applicable code classes of Section III and Section VIII of the ASME Code.

The allowable deflection limits are established by the ASME code such that the allowable stresses remain within elastic limits except where acceptance of some permanent strain is necessary to be compatible with the functional requirements.

The component and system test pressures are established in accordance with the rules for the appropriate component Class of Section III of the ASME Code.

Table 4.4 of Part 2 of the Bruce A and Bruce B Safety Reports list the operating conditions that were considered in the design. The stress analysis of all systems and major components in the HT system meets the requirements of Section III of the ASME Code. The types of stress analysis employed are tailored to the particular requirements for each system and component, and are identified in the stress reports produced for Class 1 systems and components. The faulted conditions considered in the pressure boundary analysis are identified in the stress reports produced for systems and components.

Bruce Power provisions for Level 2 for this safety principle are effective and adequate.

Level 3

The safety analysis for both Bruce A and Bruce B has demonstrated that the systems provided are capable of shutting down and maintaining the reactor subcritical following Design Basis Accidents, as well as providing adequate cooling. Any failures of internal components caused by the accident have been factored into the analyses.

As demonstrated in Part 3 of the Safety Reports for both Bruce A and Bruce B, the safety analyses have shown that for the most severe reactivity insertion accident, SDS1 can keep the reactor subcritical for at least 15 minutes, before operator action is required. This is consistent with the current CNSC requirements. SDS2 can keep the reactor shut down indefinitely without operator intervention.

The safety analyses have demonstrated that the fuel either remains cool or cooling is re-established in the event of a LOCA such that the allowable release limits are met for all AOOs and DBAs. The safety analyses have shown that even for the largest LOCA the fuel damage is limited and no failure of pressure tubes is predicted. Thus, the reactor core remains intact. In the case of a single channel failure (PT/CT rupture) the dynamic forces resulting during the blowdown cause some damage to the internal structures but enough shutoff rods remain intact to meet all the relevant requirements. The calandria vessel does not fail from the resulting over-pressure transient.

Bruce Power provisions for Level 3 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-195 Reactor core integrity

There were no strengths identified in SFRs from the standpoint of reactor core integrity.

There are 2 GIOs included in the IIP that will further improve reactor core integrity.

GIO	GIO TITLE
GIO-056	Fuel Channel Replacement
GIO-059	Calandria and Shield Tank Assembly Major Inspection

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, and 3, which are the levels applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

O-278 Training

Principle: Programmes are established for training and retraining operations and maintenance, technical support, chemistry and radiation protection personnel to enable them to perform their duties safely and efficiently. Training is particularly intensive for control room staff, and includes the use of plant simulators.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 65 for DID Levels 1-3, as described below.

Levels 1 – 3

BP-PROG-02.02 Worker Learning and Qualification ensures that all personnel involved in plant operations including operations and maintenance, technical support, chemistry and radiation protection personnel are provided with the competencies and qualifications necessary to satisfy the requirements of applicable requirements commensurate with Bruce Power business needs. The program follows the Systematic Approach to Training model defined by the INPO document: ACAD 02-001, The Objectives and Criteria for Accreditation of Training in the Nuclear Power Industry.

Bruce Power has in place training facilities, including state of the art full scope simulators used for initial certification training of Bruce Power station staff, examination of staff, and continuing training of certified staff. Bruce Power's SEC-SIMM-0001, Simulator Validation establishes the validation procedure for the full scope control room simulator. The validation procedure is used to confirm that the simulator is capable of providing the correct observable control room responses during the training and testing exercises.

Training, certification and requalification of certified personnel are conducted in accordance with the requirements in Section 2.3 Training, Certification and Examination Programs of the PROL and the accompanying LCH [5] [6].

Bruce Power provisions for Levels 1-3 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-278 Training

SF-08-S2	Furthermore, the audit organization has a well-developed Auditor Training program which used a Systematic Approach to Training based training design. Job Task Analysis is documented for knowledge and skill elements. The training program is documented and aligned to develop proficient auditors upon completion of qualifications. Auditors are professional and meet expectations of managers for performance as qualified auditors.

There is 1 GIO that will further improve training.

GIO No.	GIO TITLE
GIO-081	Human Factors in Design of Nuclear Power Plants

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, and 3, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

O-284 Operational limits and conditions

Principle: A set of operational limits and conditions is defined to identify safe boundaries for plant operation. Minimum requirements are also set for the availability of staff and equipment.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 66 for DID Levels 1-3, as described below.

Levels 1 – 3

In accordance with Licence Condition 3.1 of PROL and the accompanying LCH [5] [6] Bruce Power implements and maintains an operations program which includes:

- A Safe Operating Envelope (SOE), i.e., operational limits and conditions;
- A set of operating policies and principles; and
- Accident management procedures and/or guides for design basis accidents and accident management guides and for beyond design basis accidents, including overall strategies for recovery.

The SOE is the fundamental interface that ensures conformance between the plant design basis, safety analysis and operating documentation for safe operation of the plant in accordance with the provisions of the PROL and accompanying LCH [5] [6].

The design and the safety analysis establish an envelope of plant configurations, operating limits and conditions for safe operation. The safe operating envelope is established by defining the acceptable operating configuration, limits and conditions and incorporating these requirements in plant operating documentation. Operating documentation includes:

- Operating Policies and Principles (OP&P)
- Abnormal Incidents Manual (AIM);
- Alarm Response Manual (ARM);
- Operating Manuals (OMs);
- Overall Unit Operating Manual (09110);
- Operating Memos; and
- Safety System Testing (SST)

As part of the SOE program, if the Safety Analysis limits are adjusted, the operating documentation including items such as the Operating Manuals and Safety System Testing are adjusted to remain within a safe operating envelope.

Bruce Power has recently completed its baseline SOE project which consisted of documenting the limits and conditions derived from the safety analysis in Operational Safety Requirements (OSRs), completing the corresponding Instrument Uncertainty Calculations (IUCs), and performing Gap Assessments to verify that the requirements are completely and accurately reflected in the station operating documentation. The completion of the SOE project and subsequent programmatic activities has established a good basis for compliance with CSA N290.15 Requirements for the Safe Operating Envelope of Nuclear Power Plants, which includes the preparation of all OSRs, IUCs requirements which are consistent with the Operating Limits and Conditions (OLCs) and their basis derived from safety analysis.

Bruce Power complies with the minimum requirements set for availability of staff by strict adherence to Licence Condition 2.2 Minimum Shift Complement and Control Room Staffing of the PROL and the accompanying LCH [5] [6].

Bruce Power operates Bruce A in accordance with DIV-OPA-00001 Station Shift Complement – Bruce A, and Bruce B in accordance with DIV-OPB-00001 Station Shift Complement – Bruce B, which describe the minimum number of workers with specific qualifications required for the safe operation under all operating states and the measures in place to mitigate the impact of any minimum shift complement violations until minimum complement requirements are restored.

Bruce Power provisions for Levels 1-3 for this safety principle are effective and adequate.

<u>Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-284 Operational limits and conditions</u>

There were no strengths identified in SFRs from the standpoint of operational limits and conditions.

There are no GIOs that will further improve O-284 Operational limits and conditions. It should be noted that Bruce Power has recently completed the CSA N290.15-10 compliance project.

<u>Summary</u>

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, and 3, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

18.3.4. Safety Principles Related to Defence-in-Depth Levels 2, 3, 4

There is one 1 Safety Principle related to DID Levels 2, 3 and 4.

O-290	Emergency operating procedures

O-290 Emergency operating procedures
Principle: Emergency operating procedures are established, documented and approved to provide a basis for suitable operator response to abnormal events.

O-290 is addressed in Section 18.3.8, together with O-288 Normal Operating Procedures, as these constitute a continuum of plant operational states.

18.3.5. Safety Principles Related to Defence-in-Depth Levels 1, 2

There are 5 Safety principles related to DID Levels 1 and 2. Bruce A and Bruce B design and operation are aligned with all the safety principles as demonstrated below.

D-164	Plant process control systems
D-203	Normal heat removal
D-209	Reactor coolant system integrity
D-240	New and spent fuel storage
D-242	Physical protection of plant

D-164 Plant process control systems
Principle: Normal operation and anticipated operational occurrences are controlled so that plant and system variables remain within their operating ranges. This reduces the frequency of demands on the safety systems.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 19 and 20 for DID Levels 1 and 2, respectively, as described below.

Level 1

As part of Bruce A and Bruce B design, normal operation and upset conditions (which are similar to the modern terminology of anticipated operational occurrences) are controlled so that plant and system variables remain within their operating ranges thereby reducing the frequency of demands on the safety systems.

In addition, overpressure protection of the HT system is achieved by the combined or sole action of the reactor safety systems and fully duplicated instrumented relief valves. The RRS also acts to reduce the severity of transients but no credit is taken for its action in the safety analysis. RRS is designed to protect against AOOs leading to DBAs by invoking reactor setback or stepback functions.

Bruce A and Bruce B OP&Ps define operating requirements and licensing limits of SSCs for safe operation of the plant based on the conditions and limits set in the licence and the Safety Report. Within these operating boundaries, detailed operating procedures (i.e., OMs, Operating Memos, ARMs, SSTs, etc.) are established for clearly defined operating requirements. Normal operating procedures are prepared on how to operate the plant and abnormal operating procedures are prepared for non-routine and emergency conditions where immediate action is required. BP-PROG-12.01, Conduct of Plant Operations, and its associated documentation (Operating Procedures, ARMs, Impairments of Special Safety Systems and Other Safety Related Systems, AIMs) describe the requisite actions required to return the plant to a safe operational state.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Level 2

As part of Bruce A and Bruce B design, normal operation and upset conditions (which are similar to the modern terminology of anticipated operational occurrences) are controlled so that plant and system variables remain within their operating ranges thereby reducing the frequency of demands on the safety systems.

In addition, overpressure protection of the HT system is achieved by the combined or sole action of the reactor safety systems and fully duplicated instrumented relief valves. The RRS also acts to reduce the severity of transients but no credit is taken for its action in the safety analysis. RRS is designed to protect against AOOs leading to DBAs by invoking reactor setback or stepback functions. As part of the plant equipment protective function, automatic power reductions can be initiated via the setback or stepback functions, which are implemented in the dual, digital control computers. For many loss of control events, setback or stepback provide effective mitigating action; however such action is not credited in the safety analysis.

The setback routine is part of the reactor regulating program and monitors a number of inputs indicating the status of all setback parameters. The setback parameters are scanned every 2 s and if a parameter is out of limits and the demand power setpoint exceeds the setback endpoint, demand power is ramped down at a suitable rate until either the condition clears or the endpoint is reached. Each setback parameter may have a unique setback rate and endpoint.

The stepback routine is a computer program that runs independently of the reactor regulating program. It monitors a number of parameters, which indicate plant conditions requiring a reduction in reactor power much faster than the zone controllers can produce. If a parameter is out of limits, the program opens all four control absorber clutch contacts. If the other computer also opens its control absorber clutch contacts, the clutches will be de-energized allowing the absorbers to drop into the core. As the absorbers are dropping, the stepback routine continues monitoring the out of limits parameter, as well as extrapolated reactor power, and recloses the clutch contacts when the condition clears or extrapolated reactor power is less than the endpoint. This may result in a partial absorber drop, but most stepback conditions will cause the absorbers to be fully inserted. The stepback functions provide coverage for a variety of transients such as PHT pump trip, steam generator low level, high heat transport pressure, high zone power, high neutronic power rate, calandria inlet high temperature, turbine trip loss of line or stator cooling.

The RRS is an integrated system comprising reactor flux and thermal power measuring devices, reactivity control devices and a set of computer programs, all coordinated to perform three main functions:

- Monitor and control total reactor power so as to satisfy the station load demands.
- Monitor and control reactor flux shape.
- Monitor important plant parameters and reduce reactor power at an appropriate rate if any parameter is outside of limits.

The RRS is characterized by a high degree of immunity to small process upsets, measurement failures, etc., due to a high degree of redundancy in control devices and process measurements. Extensive checks are performed in the programs to ensure that faulty signals are discarded. In case of loss of certain signals alternative measurements are used. In case of failure of certain control devices, a backup is used. It may be necessary to derate the reactor power because of limited information or imperfect flux shape, but only as a last resort is the reactor shut down by turning off the control programs in both computers, allowing the controls to fail safe.

This ability to maintain control in the presence of partial system failures, combined with the high reliability of the dual computer control system results in a very high availability of the RRS.

Current safety analyses in the Safety Report take no credit for the control system actions. Hence, it should be noted that as required in modern codes and standards there has never been a systematic analysis of the capability of the control system to cope with AOOs (or transients in current parlance) at Bruce A and Bruce B. Some cases have been performed to demonstrate control system effectiveness for specific scenarios, usually when there was a gap in the trip coverage.

Analysis of AOOs is being addressed as part of the Safety Report Improvement activities.

Bruce Power provisions for Level 2 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-164 Plant process control systems

There were no strengths identified in SFRs from the standpoint of plant process control systems.

There are no GIOs included in the IIP that will further improve the design of plant process control systems. It should be noted that Bruce Power has recently completed the CSA N290.15-10 compliance project. In addition, analysis of AOOs is being addressed as part of the Safety Report Improvement activities under GIO-009 Update safety analysis to align with REGDOC-2.4.1.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1 and 2, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

D-203 Normal heat removal

Principle: Heat transport systems are designed for highly reliable heat removal in normal operation. They would also provide means for the removal of heat from the reactor core during anticipated operational occurrences and during most types of accidents that might occur.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 35 and 36 for DID Levels 1 and 2, respectively, as described below.

Level 1

The HT system, which carries the heat generated in the reactor core to the steam generators, is a pressurized, closed heavy water loop. The feed, bleed and relief system is designed primarily to provide a means of pressure and inventory control for this closed loop, as well as to provide adequate overpressure protection. The principal design objective for the HT system main circuit is to provide reliable cooling of the reactor fuel under all operating conditions for the life of the plant and with minimal maintenance. For the Bruce A and Bruce B designs, each material that forms a part of the reactor coolant pressure boundary has been chosen to be compatible with the expected service and environmental conditions at the location at which it is used. Low cobalt content is required for some of the major components in the HT system to keep radiation doses as low as possible. Achieving minimum leakage, maximum reliability and minimum radiation fields, providing good access for personnel and making provision for maintenance are assigned high priorities in design. The reactor coolant system is a barrier to the release of radioactive fission products and is therefore designed to retain its integrity under normal and abnormal operating conditions.

Appropriate chemistry control is maintained to minimize adverse conditions, such as reduction of heat transfer coefficients, corrosion of components, radiolytic decomposition, activation product formation, and unplanned changes in reactivity as specified in OP&P. The chemistry management program supports equipment reliability by ensuring that system chemistry is measured and controlled to design specifications. The chemistry management program, BP-PROG-12.02, interfaces with equipment reliability by providing chemistry information to plant system health reports and by providing input to lifecycle management plans, which consider the impact of chemistry-related conditions on longer-term life and ageing of components.

The pressure boundary piping is monitored periodically using non-destructive inspection techniques to assure that the likelihood of a pipe failure is kept low. In-service and periodic inspection programs including those acceptable to the CNSC provide assurance that the likelihood of in-service degradation that will lead to leaks has not increased since the plant was placed into service.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Level 2

The design of the HT circuit satisfies the rules of Section III of the ASME Code for Class 1 components.

Circulation of the reactor HT fluid is maintained at all times during reactor operation, shutdown and maintenance. In addition to the normal heat removal system, two further systems are provided for removing reactor shutdown heat, the shutdown cooling system and the maintenance cooling system. The maintenance cooling system is also designed to permit the draining of steam generators and pumps.

The Condenser Steam Discharge Valves (CSDVs) provide a means for fast and continuous rejection of the turbine steam flow, thus allowing the reactor to continue producing power at a level that will not cause a shutdown. They permit the continued operation of the nuclear steam supply system for an indefinite period in the event of a grid system or turbine generator fault. The system is designed to accept 75% of rated full power steam flow. The CSDVs are part of the steam reject/bypass system. The condenser steam discharge valves are under the control of the steam generator pressure control program in both control modes. The Atmospheric Steam Discharge Valves (ASDVs) are also part of the steam relief system and have a capacity equivalent to approximately 11% of rated full power steam flow. This capacity, together with that of the CSDVs, is sufficient to make it unnecessary to open the steam generator safety valves following most turbine trips. They also provide a means of controlling steam pressure when the CSDVs are unavailable due to poor condenser vacuum. The ASDVs are controlled by the steam generator pressure control program in both control modes.

The fuel channels, arranged so that bi-directional flow is provided in adjacent channels, are horizontal, with the headers, steam generators and pumps located above the reactor. This arrangement promotes thermo-siphoning in the event of pump failure.

Bruce Power provisions for Level 2 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-203 Normal heat removal

There were no strengths identified in SFRs from the standpoint of normal heat removal.

There are no GIOs included in the IIP that will further improve normal heat removal.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1 and 2, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

D-209 Reactor coolant system integrity

Principle: Codes and standards for nuclear vessels and piping are supplemented by additional measures to prevent conditions arising that could lead to a rupture of the primary coolant system boundary at any time during the operational lifetime of the plant.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 40 and 41 for DID Levels 1 and 2, respectively, as described below.

Level 1

The reliability of the HT system pressure boundary is assured by having applied the best available technology in design, manufacture and installation, and by making provision in design and manufacture for monitoring of the level of integrity of the pressure boundary periodically during the life of the plant.

The design of the HT circuit satisfies the rules of Section III of the ASME Code for Class 1 components. The stress analysis for piping systems and components in the HT system meets the applicable requirements of Section III of the ASME Code. The faulted conditions considered in the pressure boundary analysis are identified in the stress reports produced for systems and components.

Where appropriate, the effects of corrosion and other chemical effects such as erosion, deposition, irradiation, vibration, fire and immersion were considered in the design, and adequate design and precautionary measures were taken. Each material which forms a part of the reactor coolant pressure boundary has been chosen to be compatible with the expected service and environmental conditions at the location at which it is used.

The materials in the HT circuit meet the fracture toughness requirements for Class 1 components in Section III of the ASME Code. Generally, austenitic stainless steels are not used as part of the reactor coolant pressure boundary or in systems required for reactor shutdown or for emergency coolant injection. Wherever austenitic stainless steel is used, for instance in the gland seal systems for the HT pumps, great emphasis is given to the need for protection against contaminants during fabrication, shipment, storage, construction, testing and operation.

The reactor coolant chemistry parameters have been chosen to minimize corrosion rates on all surfaces exposed to the coolant, to minimize deposition of corrosion products on the fuel and to reduce the movement of corrosion products to an acceptably low level. In addition, operating states where conditions could lead to brittle failure are avoided, as witnessed by the limits on HT system temperature and pressure to protect pressure tube integrity. The OSRs for the HT system present the safety limits for pressure, temperature and flow, as well as surveillance requirements.

Reactor coolant system components are tested and inspected in accordance with the requirements of ASME III to assure a high quality of fabrication and installation. Inaugural and periodic inspection of the reactor coolant pressure boundary throughout the operating life of the plant assures that likelihood of failure has not increased as a result of the plant operation.

In-service degradation and ageing related OPEX from other plants is extensively used in improving in-service inspection, material surveillance, maintenance programs and fitness for service assessments to assure continued reliable operation. Feedback on operating experience is also shared with designers to improve component/material design.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Level 2

The reactor coolant chemistry parameters have been chosen to minimize corrosion rates on all surfaces exposed to the coolant, to minimize deposition of corrosion products on the fuel and to reduce the movement of corrosion products to an acceptably low level.

Precautions are taken in design to ensure that inadvertent operation of equipment will not result in unacceptable transients. This is accomplished by minimizing the number of systems and components which start up and shut down automatically and by basing the operating conditions used for component design on the worst likely combination of events.

Overpressure protection of the HT system is achieved by the combined or sole action of the reactor safety systems and fully duplicated instrumented relief valves. The reactor regulating system also acts to reduce the severity of transients but no credit is taken for its action. Protection against overpressure is designed to satisfy the requirements (except as noted in the Overpressure Protection report) for the appropriate component Class in Section III of the ASME Code.

The reactor coolant system and most auxiliaries are located within the prestressed concrete containment structure and the majority of the systems are within the normally dry reactor vault. Any leakage within this vault increases the dew point of the recirculating air and is detected. Special facilities are provided to detect moisture in the annulus gas system which may be attributed to a leak in a pressure tube. Increases in the humidity of the atmosphere in the dry vault area and in the annulus gas system are indicated and alarms are provided in the control room. Moisture detecting elements (beetles) located in each room within (and many rooms outside of) the containment structure initiate an alarm in the control room if there is water on the floor of the room.

Reactor coolant system components are tested and inspected in accordance with the requirements of ASME III to assure a high quality of fabrication and installation. Inaugural and periodic inspection of the reactor coolant pressure boundary throughout the operating life of the plant assures that likelihood of failure has not increased as a result of the plant operation.

In-service degradation and ageing related OPEX from other plants is extensively used in improving in-service inspection, material surveillance, maintenance programs and fitness-for-service assessments to assure continued reliable operation. Feedback on operating experience is also shared with designers to improve component/material design.

In summary, robust design, high quality of materials, fabrication and installation, coupled with effective use of OPEX, good chemistry control and highly sensitive leak detection systems assures leak-before-break and a very low probability of failure of the reactor coolant pressure boundary.

Bruce Power provisions for Level 2 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-209 Reactor coolant system integrity

There were no strengths identified in SFRs from the standpoint of reactor coolant system integrity.

There are 13 GIOs included in the IIP that will further improve the design of reactor coolant system integrity.

GIO No.	GIO TITLE
GIO-056	Fuel Channel Replacement
GIO-057	Steam Generator Replacement
GIO-058	Feeder Replacement
GIO-060	Preheater Inspections
GIO-065	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)- Inspection
GIO-066	Pressurizer and Supports- Internal Inspection
GIO-070	Air Operated Valves-Replacement
GIO-078	Maintenance Cooling Heat Exchanger- Replacement
GIO-086	PHT Valves-Refurbishment of 33120-MV23
GIO-099	Install Correctly Sized Maintenance Cooling Relief Valves
GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	M/34720 Relief Valves For Overpressure Protection
GIO-102	I/63472 Remote Relief Valve Position Indication

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1 and 2, which are the levels applicable to this safety principle. Thirteen GIOs were identified that will further improve the provisions for this safety principle.

D-240 New and spent fuel storage
Principle: Plant designs provide for the handling and storage of new and spent fuel in such a way as to ensure protection of workers and to prevent the release of radioactive material.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 52 for DID Levels 1-2, as described below.

Levels 1 – 2

At Bruce A and Bruce B an on-power fuelling system is provided which enables the automatic fuelling of all four reactors ensuring protection of workers. New fuel is supplied to the fuelling machine heads and irradiated fuel is discharged from the heads in the fuelling machine rooms in the Central Service Area. The discharged fuel is transferred from the heads through ports onto storage trays in the primary irradiated fuel storage bay.

Pressure boundary components of the fuelling and fuel handling systems are designed and fabricated with the applicable code class requirements of ASME III.

The design of the fuelling machine has built-in safety features for normal operations, as well as special features to deal with equipment breakdowns. For routine automatic operation, the control system provides interlocks in the stored memory computer program which are backed up in important areas by a separate protective computer system and hardwired interlocks. The machines are operated automatically, so accidents due to operator error are unlikely.

The equipment has been designed to facilitate breakdown operations and maintenance so that the personnel dose is kept to a minimum. All remote drives are arranged so that they can be manually operated. In addition, the gear box is provided with two motors and two clutches on each output shaft. Failure of one of these devices will not disable the machine. It is possible to vary the settings of the component speed and force limits remotely from the control room. Personnel exposure to radiation and tritium is thereby reduced.

The inadvertent raising of a head containing irradiated fuel above the level of the irradiated fuel port could increase the radiation levels in accessible areas, i.e., in the fuel bay and new fuel loading area in the order of 100 mGy/h (10 R/h). There is an automated check to prevent the inadvertent raising of a head. Radiation monitors also provide sufficient warning to permit affected areas to be evacuated.

The primary irradiated fuel storage bay is a reinforced concrete open-top tank, 9.65 m wide by 41.5 m long, with epoxy lined walls and a stainless steel lined floor. The bay is used for storing irradiated fuel for a minimum of six months after removal from the reactor to allow the decay heat of the bundles to subside. After the six months, the fuel may be transferred to the secondary irradiated fuel storage bay. The storage capacity of the primary bay is sufficient for 36000 fuel bundles (equivalent to ~6.25 full core load). Space is also provided for handling irradiated fuel casks, inspecting and canning defected fuel.

Irradiated fuel is stored in three areas of the primary irradiated fuel storage bay. The fuel is contained in storage trays with features at each corner for stacking. Each tray holds 24 fuel bundles in a single horizontal layer. The trays are normally stacked 15 high, providing shielding of 4.12 m of water. Under emergency conditions, it is possible to stack the storage trays up to 18 high, but this reduces the water shielding to 3.73 m.

The fuel bay water provides both coolant and radiation shielding. The fuel bay cooling circuits remove the heat generated by the fuel bundles in the bays to control the bay water temperatures for proper cooling of the fuel and to limit thermal stresses in the bay structures and the lining system. The purification circuits remove suspended and dissolved solids from the bay water to control the radioactivity level of the water for personnel protection and to maintain the clarity of the water for good visibility during inspection and transfer of the fuel bundles within the bay. Each section of the primary irradiated fuel storage bay (inspection section and storage section) and the secondary irradiated fuel storage bay have their own cooling and purification circuits.

The Bruce Used Fuel Dry Storage Project (BUFDSP) is a dry storage system based on wet-loading of used fuel into Dry Storage Containers (DSC). Fuel is transferred from the irradiated fuel trays to modules using the Tray to Module Transfer Mechanism (TMTM). The DSC is lowered into the loading bay, which is located at the north end of the secondary irradiated fuel storage bay. Four used fuel dry storage modules, containing in total 384 bundles are loaded into a DSC and the DSC is closed before removal from the bay. Decontamination of the DSC occurs during removal of the DSC from the bay. Cooling and purification of the dry fuel loading bay is provided by the secondary irradiated fuel bay systems.

The DSCs are transferred to a dry fuel storage processing area. The filled DSC is vacuum dried, filled with helium, leak tested, safeguard sealed and transferred to the Used Fuel Storage Facility on site.

New fuel is delivered to the station in crates and is stored until required for use. It is then transferred in the crates to the new fuel loading area to be loaded into the fuelling machines using the four new fuel transfer mechanisms. The new fuel transfer mechanisms transfer the fuel through the containment wall into the fuelling machines without exposing the operators to any tritium or radiation hazards. The fields in the new fuel loading area are less than 1.0×10^{-5} Gy/h (1.0 mR/h), thus allowing unrestricted access under normal conditions.

Bruce Power provisions for Levels 1 and 2 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-240 New and spent fuel storage

There were no strengths identified in SFRs from the standpoint of new and spent fuel storage.

There are no GIOs included in the IIP that will further improve new and spent fuel storage.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1 and 2, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

D-242 Physical protection of plant
Principle: The design and operation of a nuclear power plant provide adequate measures to protect the plant from damage and to prevent the unauthorized release of radioactive material arising from unauthorized acts by individuals or groups, including trespass, unauthorized diversion or removal of nuclear materials, and sabotage of the plant.

Results of Review

Security related issues design features are excluded from the PSR.

18.3.6. Safety Principles Related to Defence-in-Depth Levels 3, 4

There are 5 Safety principles related to DID Levels 3 and 4. Bruce A and Bruce B design and operation are aligned with all the safety principles as demonstrated below.

D-200	Automatic shutdown systems
D-207	Emergency heat removal
D-217	Confinement of radioactive material
D-221	Protection of confinement structure
D-233	Station blackout

D-200 Automatic shutdown systems
Principle: Rapidly responding and highly reliable reactivity reduction for safety purposes is designed to be independent of the equipment and processes used to control the reactor power. Safety shutdown action is available at all times when steps to achieve a self sustaining chain reaction are being intentionally taken or whenever a chain reaction might be initiated accidentally.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 33 and 34 for DID Levels 3-4, as described below.

Levels 3 – 4

Section 4.2.6 of Part 2 of the Bruce A and Bruce B Safety Reports describes the two fully capable, separate, independent and diverse shutdown systems. Shutdown capability is available at all times and during all phases of power operation. Each system has its own initiation sensors, detectors and logic to ensure functional and physical diversity.

Both shutdown systems are capable of shutting the reactor down fast enough for all AOOs and DBAs such that specified limits are not exceeded. There is no recriticality following accidents. For SDS1, operator action can be credited after 15 minutes to augment the depth of shutdown. For SDS2, the shutdown depth is sufficient to keep the reactor shut down indefinitely for even the most reactive conditions of the core. Bruce A and Bruce B safety analyses conservatively credit the least effective shutdown system by using the negative reactivity insertion characteristics of the slowest of the shutdown systems. SDS1 and SDS2 are functionally and physically independent and employ two diverse shutdown principles, i.e., SDS1 releases neutron absorbing spring-assisted gravity drop shutoff rods and SDS2 uses injection of a neutron absorbing solution into the moderator.

Automatic shutdown systems SDS1 and SDS2 are also described in D-192 Protection against power transient accidents.

There is a comprehensive system of monitoring, inspection, and testing to ensure the integrity of mechanical components and reliability of equipment. The development of detailed operating procedures and extensive training of plant personnel contribute to the prevention of failures in more than one SDS.

The Shutdown System OSR provides the safety limits, limiting accidents and surveillance requirements for both shutdown systems, while the surveillance frequencies are determined by the unavailability requirements for the system as confirmed by unavailability assessments.

Bruce A and Bruce B OP&Ps identify the policies and principles that drive the programs and processes for ensuring the existence of operating procedures covering normal and abnormal conditions. The OP&P is subdivided into general and specific subjects, where the front end covers multiple SSCs, records, reporting, and Section 21 onwards covers requirements for specific SSCs. For example, Section 10.4 of the OP&P mentions that abnormal or emergency condition procedures written to protect the public and station personnel in emergencies involving the release of radioactive material, emergency procedures shall be implemented specifying staff responsibilities, available equipment, prerequisite training, and procedures to be followed. Duties of responsible individuals during normal and abnormal operation are defined in Section 01 of the OP&P.

BP-PROG-12.01, Conduct of Plant Operations covers operations documentation and plant operation for normal and abnormal and emergency conditions. Normal operating procedures are written for a wide range of systems and situations. Operating procedures include AIMs, OMs, Operating Memos, ARMs, and Safety System test procedures. The Overall Unit Operating Manual covers standard and non-standard operating conditions. Procedures for the safe and reliable operation of plant equipment are prepared, approved, controlled and readily available to the operating staff. These procedures are prepared for anticipated normal, abnormal and emergency conditions.

Bruce Power provisions for Levels 3 and 4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-200 Automatic shutdown systems

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There are 4 GIOs that will further improve D-200 automatic shutdown systems.

GIO No.	GIO TITLE
GIO-026	BA & BB New Neutronic Trips
GIO-076	DCC Cables and WIBAs –Replacement
GIO-090	SDS2 Enhancements
GIO-095	45VDC Power Supplies-Replacement

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 3 and 4, which are the levels applicable to this safety principle. Four GIOs were identified that will further improve the provisions for this safety principle.

D-207 Emergency heat removal

Principle: Provision is made for alternative means to restore and maintain fuel cooling under accident conditions, even if normal heat removal fails or the integrity of the primary cooling system boundary is lost.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 38 and 39 for DID Levels 3 and 4, respectively, as described below.

Level 3

The ECI system is a special safety system that has no role during normal operation. The ECI system refills the HT system and keeps it filled, if required, under accident conditions such as a large break LOCA. The system provides a long-term heat sink for emergency core cooling. It is designed to operate under post-LOCA conditions.

The ECI system in each of Bruce A and B is common to all four units. A 76 cm (30 in) diameter common supply header runs the length of the station. The header is thermally insulated as required to reduce heat input to the header from secondary side failures. Injection lines to each individual unit contain a parallel pair of normally closed motorized water injection valves, outside the containment structure. An inverted U-bend provides an air gap, which forms an interface between the light water and heavy water systems. Four branch lines then penetrate the containment structure providing for injection to four quadrants of the HT system.

The ECI system, which is inactive but poised during normal operation of the station, is activated automatically when a loss of coolant accident is detected in any unit. An emergency coolant injection signal is initiated when the HT pressure falls below a set value in conjunction with another parameter that indicates a LOCA, such as high reactor vault pressure or if the HT system remains below 5.5 MPa for an extended time period.

In addition, the Emergency Boiler Cooling (EBC) system at Bruce A and Emergency Water System (EWS) at Bruce B provide feedwater to steam generators to ensure adequate decay heat removal in the event of a main steam line break resulting in the loss of normal feed. As part of the Fukushima Follow-up Actions, design provisions for external water makeup to the HT and Moderator systems are being implemented to improve severe accident response.

Plant capabilities for challenges posed by DBAs (and some BDBAs) are assessed and confirmed within appropriate limits through analyses documented in the Safety Report. Furthermore, the PRA confirms that the safety goals are met. Emergency Operating Procedures (EOPs), SAMG, and Emergency Plan actions reduce risks from possible releases of radioactivity. The EOPs and AIMs address DBAs regardless of the initiating hazard. Accident management measures are identified and implemented through site-specific OMs, AIMs, SAMG Emergency Response Procedures and Emergency Mitigating Equipment Guidance (EMEG) to ensure adequate capabilities are maintained to cope with scenarios ranging from AOOs to severe accidents. Specific OMs and AIMs cover shutdown states and accidents involving the IFB. Updates of the SAMGs to account for multi-unit events, hydrogen management, in-vessel retention, and IFB are complete.

Bruce Power provisions for Level 3 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Level 4

The ECI system is a special safety system that has no role during normal operation. The ECI system refills the HT system and keeps it filled, if required, under accident conditions such as a large break LOCA. The system provides a long-term heat sink for emergency core cooling. It is designed to operate under post-LOCA conditions.

The ECI system, which is inactive but poised during normal operation of the station, is activated automatically when a loss of coolant accident is detected in any unit. An emergency coolant injection signal is initiated when the HT pressure falls below a set value in conjunction with another parameter that indicates a LOCA, such as high reactor vault pressure or if the HT system remains below 5.5 MPa for an extended time period.

In addition, the EBC system at Bruce A and EWS at Bruce B provide feedwater to steam generators to ensure adequate decay heat removal in the event of a main steam line break resulting in the loss of normal feed. As part of the Fukushima Follow-up Actions, design provisions for external water makeup to the HT and Moderator systems are being implemented to improve severe accident response.

Bruce Power provisions for Level 4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-207 Emergency heat removal

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There are 2 GIOs included in the IIP that will further improve emergency heat removal.

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response
GIO-070	Air Operated Valves-Replacement

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 3 and 4, which are the levels applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

D-217 Confinement of radioactive material

Principle: The plant is designed to be capable of retaining the bulk of the radioactive material that might be released from fuel, for the entire range of accidents considered in the design.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 42 and 43 for DID Level 3 and Figure 44 for Level 4, as described below.

Level 3

In both Bruce A and Bruce B, Containment is a special safety system that forms an envelope around the nuclear components of the reactor and the reactor coolant system. It consists of a number of systems and subsystems whose collective purpose is to prevent any significant release of radionuclides, which may be present in the containment atmosphere following certain postulated accident conditions, to the outside environment. The physical barrier, which minimizes the outflow of radionuclides, is called the containment envelope. An important criterion for determining the effectiveness of the containment envelope is the integrated leak rate for the period of the pressure excursion. To meet the design leakage requirements, two measures are employed. The first involves stringent design requirements to minimize the leak rate. The second is to prevent the design pressure within the containment envelope from being exceeded following a LOCA. The containment system quickly reduces the containment pressure pulse to a sub-atmospheric level following a large energy release within the containment envelope and hence minimizes uncontrolled releases to the outside environment.

The containment envelope includes the four reactor vaults, the fuelling duct, the central fuelling area, the east service area, the pressure relief ducts, the pressure relief valve manifold, the vacuum building, airlocks and transfer chambers, and extensions of containment arising from numerous piping penetrations. The majority of the extensions are normally closed and a number are normally open. The normally open extensions are automatically closed following the detection of high activity or high pressure inside containment thus ensuring that a closed envelope is provided to contain potential activity in the event of an accident.

The pressure is normally maintained at 6.9 to 10.3 kPa absolute (1.0 to 1.5 psia) in the vacuum building, and at slightly subatmospheric in the rest of the containment envelope (-2.5 kPa to -3.5 kPa(g) at Bruce A and (-2 kPa to -3 kPa(g) at Bruce B).

Operation of the containment pressure suppression system is automatic and passive in nature. Consequently, few control systems directly related to the containment function are required. The pressure relief valves are actuated by a rise in pressure in the pressure relief duct, and the dousing spray system in the vacuum building is actuated by a rise in the vacuum building pressure. The dousing water spray system consists of an emergency water storage tank in the top of the vacuum building and a system of spray headers. The function of the dousing system is to condense any steam discharged into the vacuum building, to cool the steam and air mixture in the building and thus limit any pressure rise. Thus, the energy released by the accident actuates these safety devices. All systems connected to the containment atmosphere are provided with adequate barriers which automatically isolate following an accident. Either a high containment pressure signal or a high radioactivity indication initiates this containment isolation. Personnel access to the containment envelope is by means of airlocks or transfer chambers to ensure the integrity of the envelope.

The various components of the containment system can be tested separately to demonstrate the integrity of the components, as well as the system as a whole. A constant check on the leakage in the vacuum building is observed from the operation of the vacuum pumps. Pressure indications are also available for monitoring containment leakage. In addition, tests are conducted on a quarterly basis to permit an estimate of leakage into the reactor buildings and the fuelling machine duct, excluding the vacuum building.

Periodic inspection of containment boundary metallic components, concrete structures and positive pressure testing of the same is also performed at regular intervals assuring leak tightness of the containment is within its design and operating envelope.

Plant capabilities for challenges posed by DBAs (and some BDBAs) are assessed and confirmed within appropriate limits through analyses documented in the Safety Report. Furthermore, the PRA confirms that the safety goals are met.

Bruce Power provisions for Level 3 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Level 4

Most of the Level 3 provisions discussed above also apply to Level 4. Only those specific to Level 4 are discussed below.

EFADS is operated to control long-term radiological dose to the public and station staff by providing a well-defined, filtered, controlled and monitored release path of fission products from containment following a LOCA or other DBAs. The system consists of two 100% filters and blowers plus duct work and isolation dampers. Each filter contains a demister, heater, prefilter, upstream HEPA filter, charcoal filter and downstream HEPA filter. The exhaust flow is drawn from the vacuum building or the pressure relief valve manifold and is monitored by the post-accident radiation monitoring system prior to being released to the atmosphere via the system exhaust stack. A recirculation line enables pre-discharge monitoring of the exhaust flow prior to the end of the subatmospheric hold up period. An alternative exhaust path from the pressure relief valve manifold also is available.

PARMS provides on-line radioisotopic analysis for noble gases, gross gamma detection and off-line radioisotopic analyses for particulates, iodine and tritium. The detected and analyzed parameters are presented on a local and a remote display unit, located in the Unit 2 control equipment room.

Plant capabilities for challenges posed by DBAs (and some BDBAs) are assessed and confirmed within appropriate limits through analyses documented in the Safety Report. Furthermore, the PRA confirms that the safety goals are met. EOPs, SAMG, and Emergency Plan actions reduce risks from possible releases of radioactivity. The EOPs and AIMs address DBAs regardless of the hazard initiating the DBA. Accident management measures are identified and implemented through site-specific OMs, AIMs, SAMG Emergency Response Procedures and EMEG to ensure adequate capabilities are maintained to cope with scenarios ranging from AOOs to severe accidents. Specific OMs and AIMs cover shutdown states and accidents involving the Irradiated Fuel Bay. Updates of the SAMGs to account for multi-unit events, hydrogen management, in-vessel retention, and IFB are complete.

In addition to the provisions in place for DBAs, complementary design features and operational enhancements are being implemented that ensure effectiveness of the containment function during severe accidents through Bruce Power's response to the CNSC Action Plan on Fukushima Action Items. These improvements include design provisions for a containment connection, as well as the installation of a passive CFVS.

Bruce Power provisions for Level 4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-217 Confinement of radioactive material

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There is 1 GIO that will further improve confinement of radioactive material

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 3 and 4, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

D-221 Protection of confinement structure

Principle: If specific and inherent features of a nuclear power plant would not prevent detrimental effects on the confinement structure in a severe accident, special protection against the effects of such accidents is provided, to the extent needed to meet the general safety objective.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 45 and 46 for DID Levels 3 and 4, respectively, as described below.

Level 3

In both Bruce A and Bruce B, Containment is a special safety system that forms an envelope around the nuclear components of the reactor and the reactor coolant system. It consists of a number of systems and subsystems whose collective purpose is to prevent any significant release of radionuclides, which may be present in the containment atmosphere following certain postulated accident conditions, to the outside environment. The containment system quickly reduces the containment pressure pulse to a sub-atmospheric level following a large energy release within the containment envelope, and hence minimizes uncontrolled releases to the outside environment.

The containment structures were subjected to the positive and negative proof test pressures to confirm the structural integrity of containment. The overall containment integrity is confirmed by a positive pressure test of the entire system, during station outages. Containment performance is also monitored and trended via the quarterly on-power leak rate test, which measures the leak tightness of the containment structure at negative pressure.

The design provides for automatic containment pressure suppression that is predominantly passive. The pressure relief valves are actuated by a rise in pressure in the pressure relief duct, and the dousing spray system in the vacuum building is actuated by a rise in the vacuum building pressure. The dousing water spray system consists of an emergency water storage tank in the top of the vacuum building and a system of spray headers. The function of the dousing system is to condense any steam discharged into the vacuum building, to cool the steam and air mixture in the building and thus limit any pressure rise.

Features incorporated into the Bruce A and Bruce B designs provide an adequate level of protection against any credible turbine generator missile. These include separation of the 600 V

Class II switchgear, reinforced concrete barriers, and adoption of separation measures, such that a single missile cannot disable sufficient equipment to prevent safe shutdown, monitoring, or decay heat removal.

Heat removal from containment is provided by an air-to-water cooling system. The vault cooling system performs a long-term containment function following a LOCA by providing sufficient heat removal capacity to assist in maintaining the integrity of the containment envelope.

Plant capabilities for challenges posed by DBAs (and some BDBAs) are assessed and confirmed within appropriate limits through analyses documented in the Safety Report. Furthermore, the PRA confirms that the safety goals are met.

At both Bruce A and Bruce B, two separate systems are provided for mitigation of hydrogen following the low probability design basis event combinations and BDBAs.

- Hydrogen Ignition System for mitigation of short term hydrogen generation, and
- Passive Autocatalytic Recombiners (PARs) for slower longer term hydrogen generation such as from radiolysis of water. PARs provide defence-in-depth for short term hydrogen mitigation as well.

Bruce Power provisions for Level 3 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Level 4

Most of the Level 3 provisions discussed above also apply to Level 4. Only those specific to Level 4 are discussed below.

EFADS is operated to control long-term radiological dose to the public and station staff by providing a well-defined, filtered, controlled and monitored release path of fission products from containment following a LOCA or other DBAs. The exhaust flow is drawn from the vacuum building or the pressure relief valve manifold and is monitored by the post-accident radiation monitoring system prior to being released to the atmosphere via the system exhaust stack.

Heat removal from containment is provided by an air-to-water cooling system. The vault cooling system performs a long-term containment function following a LOCA by providing sufficient heat removal capacity to assist in maintaining the integrity of the containment envelope.

The addition of water to cool the fuel debris can create consequential challenges to containment, specifically overpressurization due to the production of steam, increased hydrogen generation, and the buildup of water level on the containment floor. The in-vessel retention strategy aims to prevent corium concrete interactions (as a result of subsequent calandria vault / shield tank failure), which reduces much of the uncertainty with respect to maintaining containment integrity and represents a success of mitigating actions to recover control in the event of a severe accident. The research documents from COG JP 4426, CANDU Severe Accident Support to Industry – Post Fukushima concluded that “the combination of existing plant features in supporting analyses (e.g., Level 2 PSA) and various plant enhancements, either planned or under active evaluation by the utilities a part of their post-Fukushima response, provide confidence that maintaining containment integrity is an achievable goal following a severe accident.”

At both Bruce A and Bruce B, two separate systems are provided for mitigation of hydrogen following the low probability design basis event combinations and BDBAs.

- Hydrogen Ignition System for mitigation of short term hydrogen generation, and
- PARs for slower longer term hydrogen generation such as from radiolysis of water. PARs provide defence-in-depth for short term hydrogen mitigation as well.

In addition to the provisions in place for DBAs, complementary design features and operational enhancements are being implemented that ensure effectiveness of the containment function during severe accidents through Bruce Power's response to the CNSC Action Plan on Fukushima Action Items. These improvements include design provisions for a containment connection, as well as the installation of a passive CFVS.

Bruce Power provisions for Level 4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-221 Protection of confinement structure

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There is 1 GIO that will further improve protection of confinement structure.

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 3 and 4, which are the levels applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

D-233 Station blackout
Principle: Nuclear plants are so designed that the simultaneous loss of onsite and offsite AC electrical power (a station blackout) will not soon lead to fuel damage. The use of 'simultaneous' is not intended to imply that the loss of onsite and offsite power necessarily occurs at the same time.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 50 for DID Levels 3-4, as described below.

Levels 3 – 4

Provisions for mitigating complete loss of onsite and offsite AC power were not considered in the original design of Bruce A and Bruce B electrical power systems. Since the heat transport system pumps are one of the major unit Class IV system loads failures in the Class IV power system can result in a loss of power to one or more of these pumps, with a consequent reduction of forced circulation in the heat transport system. The safety concerns associated with such events are possible impairment of fuel cooling capability and pressurization of the heat transport system which may pose a threat to the integrity of the heat transport system. Analysis of a number of postulated failures in the Class IV power system, leading to either total or partial loss of Class IV power to a unit is performed to demonstrate the capability of the design to accommodate such failures. The current safety analysis as documented in Part 3 of the Safety Report does not consider events with station blackout. However, such events will be addressed as part of the Safety Report Improvement Project.

Electrical modifications to allow the quick connection of portable generators to backfeed into the Qualified Power Supply (QPS) at Bruce A and into the Emergency Power Supply (EPS) at Bruce B were previously completed in 2012. This modification allows key instrumentation and control equipment to remain operable for an indefinite period of time. Procurement of Emergency Mitigating Equipment (EME) (fire trucks, portable generators, refuelling truck, portable pumps, etc.) has been completed. The SAMG will address multi-unit events involving a station blackout.

Bruce Power provisions for Levels 3 and 4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-233 Station blackout

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There are no additional planned initiatives included in the IIP that will further improve station blackout.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 3 and 4, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

18.3.7. Safety Principles Related to Defence-in-Depth Levels 4, 5

There are 2 Safety principles related to DID Levels 4 and 5. Bruce A and Bruce B design and operation are aligned with all the safety principles as demonstrated below.

EP-333	Emergency plans
EP-336	Emergency response facilities

EP-333 Emergency plans
Principle: Emergency plans are prepared before the startup of the plant, and are exercised periodically to ensure that protection measures can be implemented in the event of an accident which results in, or has the potential for, significant releases of radioactive materials within and beyond the site boundary. Emergency planning zones defined around the plant allow for the use of a graded response.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 77 and 78 for DID Levels 4 and 5, respectively, as described below.

Level 4

Bruce Power's Nuclear Emergency Response Plan (NERP), BP-PLAN-00001 is referenced in the LCH and is subject to document version control such that changes to Bruce Power's Nuclear Emergency Response Plan require notification to the Commission, or a person authorized by the Commission, prior to implementation.

Bruce Power's Nuclear Emergency Response Plan and the supporting site-specific procedures listed in Appendix A to the Nuclear Emergency Response Plan include:

- On-going review of corporate risks (conducted a minimum of every five years) to determine planning requirements;
- A planning basis that, in addition to DBAs, takes into account requirements to support a sustained response to a Beyond Design Basis multi-unit event resulting in an extended loss of off-site power for up to 72 hours without assistance;
- The designation of persons for directing on-site activities and for ensuring liaison with off-site organizations;
- The conditions under which an emergency shall be declared, a list of job titles and/or functions of persons empowered to declare it, and a description of suitable means for alerting response personnel and public authorities;
- The arrangements for initial and subsequent assessment of the radiological conditions on and off the site;
- Provisions for minimizing the exposure of persons to ionizing radiation and for ensuring medical treatment of casualties;

- Assessment of the state of the installation and the actions to be taken on the site to limit the extent of radioactive release;
- The chain of command and communication, including a description of related facilities and procedures;
- An inventory of the emergency equipment to be kept in readiness at specified locations;
- The actions to be taken by persons and organizations involved in the implementation of the plan; and
- Provisions for declaring the termination of an emergency.

The emergency response plan maintenance requirements are defined in Section 4.1.3 of the Bruce Power Nuclear Emergency Response Plan. BP-PROG-08.01, Emergency Measures Program performance is assessed in relation to its purpose using the criteria found in BP-PROC-00010, Emergency Preparedness Drills and Exercises. These include a variety of review and assessment mechanisms as further defined by implementing procedures, including drills and exercises, administrative requirements management, and program assessment (which includes quality assurance assessments, self-assessments, and independent assessments). Program Assessment results are reported to Bruce Power Management and corrective actions developed and implemented to address those gaps, if required. These processes, in conjunction with planning basis review processes, OPEX, and external jurisdiction reviews provide regular assessments of the adequacy and need for updating of emergency plans and procedures. Also, per BP-PROC-00166 General Procedure and Process Requirements, BP-PLAN-00001 Nuclear Emergency Response Plan is subject to periodic reviews through Action Requests.

Bruce Power is also implementing a transition plan for REGDOC-2.10.1 Nuclear Emergency Preparedness and Response in accordance with Section 10.1 of the LCH. A detailed gap analysis has been completed and a transition plan has been developed to close the identified gaps. The key milestones of the transition plan, some of which are more applicable to Level 5, are as follows:

- Develop a Bruce Power Recovery Plan
- Complete the On-Site/Off-Site Emergency Response Communications Project to ensure that two independent means of communication are available to all emergency centres.
- Update the Bruce Emergency Response Code to predict off-site radiation dose to the public for severe and multi-unit accident scenarios.
- Complete KI pill pre-distribution out to 50 kilometers. (This is complete, per Item 129 of the Minutes of the CNSC Meeting held September 30 and October 1, 2015 [35].)
- Establish a contract to complete public evacuation time estimates.

Bruce Power will be in full compliance with REGDOC-2.10.1 *Nuclear Emergency Preparedness Response* by August 31, 2018 per Section 10.1 of the LCH [6].

Level 5

Bruce Power's Nuclear Emergency Response Plan, BP-PLAN-00001 and its supporting documentation discussed for the Level 4 defence are also applicable to the Level 5 defence.

Bruce Power's Nuclear Emergency Response Plan, BP-PLAN-00001 and supporting documentation are in place. The plan stipulates periodic exercise to ensure that protection measures can be implemented in the event of an accident. Bruce Power provisions for Levels 4 and 5 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

The emergency plan applies to both Level 4 and Level 5 defences and, therefore, does not, in itself provide independent defence. However, independence of defence-in-depth provisions is achieved by the operator and technical support staff training for response to emergencies, the roles of the emergency support centres, and in the procedures subordinate to the NERP.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to EP-333 Emergency plans

SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There are 2 GIOs that will further improve emergency plans.

GIO No.	GIO TITLE
GIO-044	Emergency preparedness
GIO-089	Whole-Site Probabilistic Risk Assessment

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 4 and 5, which are the levels applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

EP-336 Emergency response facilities

Principle: A permanently equipped emergency centre is available off the site for emergency response. On the site, a similar centre is provided for directing emergency activities within the plant and communicating with the off-site emergency organization.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figures 77 and 78 for DID Levels 4 and 5, respectively, as described below.

Level 4

Bruce Power's Nuclear Emergency Response Plan, BP-PLAN-00001 addresses the on-site technical and operations support centre, i.e., the Main Control room, and the Emergency Operations Centre (EOC), and the off-site Emergency Management Centre (EMC). The emergency response plan maintenance requirements are defined in Section 4.1.3 of BP-PLAN-00001, Bruce Power Nuclear Emergency Response Plan. These include maintenance and testing of equipment and facilities, which provide on-going assurance of the adequacy of these emergency centres. As indicated in Section 5.3.1 of BP-PLAN-00001, Bruce Power Nuclear Emergency Response Plan, facilities and equipment are being maintained on a routine basis.

The EMC includes a back-up power supply to ensure the EMC is capable of providing continuous AC power to critical building loads and equipment for at least 72 hours after a BDBA. Emergency response crews have been trained and exercised multiple times in the deployment of the emergency equipment.

As a result of Fukushima Action Item completion, communications upgrades have been completed, including a radio communications infrastructure and satellite phone capability both at the EMC and the Central Maintenance and Laundry Facility (CMLF). Further enhancements included the installation of a VSAT (Very Small Aperture Terminal) system at the EMC to provide multiple backup phone hubs and internet connectivity. These upgrades address connectivity issues between the EMC and station EOC, as well as external agencies.

Level 5

The same defences described for Level 4 apply to Level 5, since the emergency support centres from which control of severe accident conditions occurs are the same as those for mitigating releases of radioactive materials after a damaged reactor has been placed into a stable state.

The on-site technical and operations support centres, i.e., the Main Control Room, and the Emergency Operations Centre, are equipped with the necessary communications and other equipment as described in Section 4.1.2.2 of BP-PLAN-00001, Bruce Power Nuclear Emergency Response Plan.

Bruce Power has recently consolidated the Site Management Centre and the Corporate Emergency Support Centre into an Emergency Management Centre located at the Bruce Power Visitor's Centre. This facility is located outside the site boundary in order to improve arrangements, including supporting the Incident Management System, which is also used by the Provincial Emergency Operations Centre.

The emergency technical and operations support centres are in place with secure communications networks. Bruce Power provisions for Levels 4 and 5 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

The support centres apply to both Level 4 and Level 5 defences and, therefore, do not, in themselves provide independent defences. However, independence of defence-in-depth provisions is achieved by the operator and technical support staff training for response to emergencies and the roles of the emergency support centres.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to EP-336 Emergency response facilities

SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There are no additional planned initiatives included in the IIP that will further improve emergency response facilities.

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 4 and 5, which are the levels applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

18.3.8. Safety Principles Related to Defence-in-Depth Level 1

There are 4 Safety principles related to DID Level 1. Bruce A and Bruce B design and operation is aligned with all the safety principles as demonstrated below.

S-136	External factors affecting the plant
D-188	Radiation protection in design
O-272	Conduct of operations
O-288	Normal operating procedures

S-136 External factors affecting the plant
Principle: The choice of site takes into account the results of investigations of local factors that could adversely affect the safety of the plant.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 11 for DID Level 1, as described below.

Level 1

The characteristics of the site for Bruce A and Bruce B are included in the Safety Reports Part 1:

- Geography – Including topography, site access, population, agriculture, industry, transportation, fishing and recreation
- Meteorology – Including severe meteorological conditions, regional climatology, temperature, precipitation and lake effect

- Hydrology – Including lake water (currents, wave heights, water levels), water temperatures (thermal plumes and ground water)
- Geology and Seismology – Including regional seismicity, seismic ground motion and seismic design of nuclear structures

In addition, the ongoing concern with climate change due to human impact on the environment requires some consideration with respect to the effect on expected severe weather conditions for the area around the Bruce Power Site. The temperature data gathered indicates that the temperature near Bruce NGS site is increasing. Climate Canada suggests that Ontario could experience anywhere from 3 – 8ºC average annual warming by the latter part of the 21st century, leading to fewer weeks of snow, a longer growing season, less moisture in the soil, and an increase in the frequency and severity of droughts. Increased atmospheric temperatures are expected to lead to an increase in the temperature of the water in the Great Lakes. This has a potential to impact on the ability of the lake to supply cooling water to the plant. Bruce A and Bruce B water intake temperature is monitored daily and currently there is no indication of an increasing trend in these temperatures. Bruce Power will continue to monitor for any adverse trends.

Bruce Power undertook, as part of its disposition of Fukushima Action Items, a re-evaluation of the site-specific magnitudes of external events to which the plant might be susceptible, using modern calculations and methods; and an evaluation as to whether the current site-specific design protection for each external event so assessed is sufficient. An extensive screening assessment was conducted based on a screening methodology submitted to CNSC.

These hazards were initially subjected to a first-level screening, and the hazards which were not eliminated in the first level were then subjected to a second level of screening. Following this second level of screening, the hazards requiring further assessment are tornados, high winds and external flooding. Bruce Power has also submitted a methodology for analysis of tornados, high winds and external flooding, and more recently for both Bruce A and Bruce B High Wind PRA Report and External Flood Assessment, as well as a Seismic PRA Report and Fire PRA Report.

The Bruce A and Bruce B Safety Reports address local factors in its determination of safety of the site. Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to S-136 External factors affecting the plants

There were no strengths identified in SFRs from the standpoint of external factors affecting the plant.

There are 3 GIOs included in the IIP that will further improve external factors affecting the plant.

GIO No.	GIO TITLE
GIO-009	Update safety analysis to align with REGDOC-2.4.1
GIO-083	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2

GIO No.	GIO TITLE
GIO-089	Whole-Site Probabilistic Risk Assessment

Summary

There are effective and adequate provisions in place for defence-in-depth Level 1, which is the level applicable to this safety principle. Three GIOs were identified that will further improve the provisions for this safety principle.

D-188 Radiation protection in design

Principle: At the design stage, radiation protection features are incorporated to protect plant personnel from radiation exposure and to keep emissions of radioactive effluents within prescribed limits.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 26 for DID Level 1, as described below.

Level 1

The design provisions for radiation protection include appropriate shielding, filtration, venting and sampling in order to limit the exposure of plant personnel as low as reasonably achievable. BP-PROG-12.05 Radiation Protection Program is in place to support this goal.

As described in Part 2, Section 12.2 of the Bruce A and Bruce B Safety Reports, all systems considered to have significant radiological implications for station personnel during operation or maintenance were reviewed in the design phase. The review process included a series of Man-Rem Audit meetings on a system-by-system basis. AECL design, operations, health physics, and physics and analysis groups were represented. Each system design was examined with respect to reliability, maintainability, ease of handling, ease of access, shielding, etc. Radiation exposure was estimated for each system in Man-Rem per year, and the estimate compared with budgeted exposure figures prepared earlier as targets. (All estimates were based on Douglas Point radiation exposure data as reported for 1970). Proposals to reduce radiation exposure by improving system design were analyzed and, wherever feasible, implemented. Special attention was also directed to system chemistry, equipment simplicity, service intervals, and ease of component removal. In general, it was recognized that the fundamental approach of improving component reliability or system chemistry is more effective than secondary measures such as installation of additional shielding. Improved station design has contributed significantly to the reduction of both collective and individual dose expenditures, and to the productivity of those dose expenditures which do take place.

Limiting personnel exposure is achieved by incorporating protective features into the initial station design, by controlling access to areas with elevated radiation levels, and by excluding personnel who are approaching certain administrative dose limits from further exposure. Requirements are in place that govern the use of Radiation Protection Protective Equipment, which protect personnel from internal radiation resulting from the uptake of airborne and surface contamination. Decontamination facilities are provided to restrict the spread of contamination.

Dosimetry and personnel monitoring devices are used extensively to monitor the doses that staff members receive, and to ensure that these doses are within allowable limits.

The plant is laid out to minimize the need for personnel to enter areas with high radiation fields. In general, operational procedures restrict access to the reactor building to qualified personnel and those escorted by qualified personnel. Access to areas that either have or could have high radiation fields is strictly controlled by the Access Control System. Extensive use is made of physical barriers, permanent and temporary signs, and other means to clearly warn and instruct personnel of any possible danger from radiation.

The station is divided into three zones according to the potential for contamination and other radiological hazards. For any movement of personnel or material between zones, actions must be taken to prevent possible contamination from a zone of higher number to a zone of lower number. For this purpose, contamination monitors are located on all approved routes between zones.

BP-RPP-00015 Zoning details the requirements for movement of personnel and equipment around the zoned areas of Bruce Power Facilities and specifies the requirements for the transfer of radioactive material outside the zoned areas but within the site boundary. The contamination limits for Zone 1 and Unzoned area surfaces are presented in Appendix A of BP-RPP-00015 Zoning.

There are numerous decontamination centres within the plant, located at appropriate locations, to handle contaminated equipment, e.g., Fuelling Machine Dismantling and Decontamination Room, Small Parts Decontamination Room, fuel shipping cask decontamination area is provided in the shipping area, CSA decontamination facilities, etc.

The Bruce Power Design Standard, Radionuclide Effluent Monitoring System Requirements, B-ST-03480-10000, provides detailed guidance on the requirements for performance and control monitoring of airborne and waterborne effluent streams. In this document, “performance monitoring” is defined as “the monitoring of an emission source that potentially could emit an amount of radioactivity equivalent to a significant proportion of any Derived Release Limit. Performance monitoring is required to demonstrate compliance with regulatory limits, measure emissions performance, calculate the potential dose impact to a critical group.” Control monitoring is defined as, “the monitoring of an emission source to provide adequate warning to ensure automatic or operator action can be taken so targets and regulatory limits are not exceeded,” and is equivalent to the process monitoring defined in N288.5-11. Control monitoring sampling frequencies for continuous streams are specified to ensure that no more than 5% of the applicable weekly DRLs could be released without detection and alarm.

Station layout, zoning, shielding, filtering, venting and sampling have been addressed in the station design to minimize personnel exposure to radioactive contamination and radiation fields. The design requirements for site and off-site monitoring of emissions provides adequate warning to ensure that emission targets and regulatory limits are not exceeded. Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-188 Radiation protection in design

There were no strengths identified in SFRs from the standpoint of radiation protection in design.

There are 2 GIOs that will further improve radiation protection in design.

GIO No.	GIO TITLE
GIO-037	Document design basis for zoning and shielding
GIO-082	Performance testing of nuclear air-cleaning systems

Summary

There are effective and adequate provisions in place for defence-in-depth Level 1, which is the level applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

O-272 Conduct of operations
Principle: Operation of the plant is conducted by authorized personnel, according to strict administrative controls and observing procedural discipline.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 64 for DID Level 1, as described below.

Level 1

The overall objective of BP-PROG-12.01 Conduct of Plant Operations is to safely and reliably operate the station systems within the design basis for which the plants are licensed. DIV-OPA-00001 Station Shift Complement- Bruce A and DIV-OPB-00001 Station Shift Complement - Bruce B describe the minimum number of workers with specific qualifications required for the safe operation of the nuclear facilities under all operating states and the measures in place to mitigate the impact of any minimum shift complement violations until minimum complement requirements are restored. BP-PROG-12.01 Conduct of Plant Operations and its implementing procedures describe administrative and procedural requirements and adherence to the same to ensure safe operation of the plant in accordance with the provisions of the PROL and the accompanying LCH [5] [6].

Operations conducted in accordance with the standards and expectations defined in BP-PROG-12.01 Conduct of Plant Operations provide strong support for the four pillars of nuclear safety: reactor safety; industrial safety; radiological safety; and environmental safety.

The four operational areas implemented by the Conduct of Plant Operations program are:

- Operations Documentation - Controls the development, review, and approval of all procedures, flowsheets, and other documents used by Operations personnel.

- Operator Staffing - Controls the activities to ensure qualified Operations staff complements are acceptable for the safe operation of the reactor units and for the performance of routine and outage activities.
- Plant Operation - Controls the execution of Operator activities in the plants to start-up, operate and shut down the reactor units, to refuel the reactors on an on-going basis, to perform routine operations in support of maintenance activities, and to perform routine surveillance of systems and to respond to unanticipated events.
- Work Protection - Controls the development and approval of Work Protection related procedures and oversees the execution of Work Protection related activities to ensure an isolated and de-energized condition exists for the execution of work.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

<u>Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-272 Conduct of operations</u>

SF-02-S2	Bruce Power's preventive maintenance implementation is a station priority. The station management team monitors implementation and leaders enforce accountability
SF-04-S1	Information from the Asset Management Program is proactively used to inform the business of the future needs related to ageing and to ensure the funding and priorities can be proactively established as required to ensure effective ageing management and plant safety.
SF-04-S2	Bruce B is an industry leader in the area of managing obsolescence of technology as evidenced by being awarded a WANO Strength and being the subject of a WANO Good Practice publication
SF-08-S1	An observed strength involves the commitments to improvements that are systematically being undertaken, based on the strong direction and guidance from the Nuclear Oversight and Regulatory Affairs organization, both in their audit and assessment reviews and their push to comply with more recent Regulatory Documents, Guidance Documents and Standards. The organization was re-organized to improve their focus on both Audits and Assessments and has committed to the CNSC to introduce a risk-informed process to their audits and assessments process to ensure risk significant areas are reviewed more frequently. (Same strength observed as in SF-10-S2 and SF-11-S2)

There are no additional planned initiatives included in the IIP that will further improve conduct of operations.

<u>Summary</u>

There are effective and adequate provisions in place for defence-in-depth Level 1, which is the level applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

O-288 Normal operating procedures
Principle: Normal plant operation is controlled by detailed, validated and formally approved procedures.

O-290 Emergency operating procedures
Principle: Emergency operating procedures are established, documented and approved to provide a basis for suitable operator response to abnormal events.

Results of Review

Both O-288 and O-290 are addressed below as they constitute a continuum of plant operational states.

These safety principles are mapped to SRS-46 Objective Tree Figure 67 for DID Level 1 and Figure 68 for DID Levels 2-4, as described below.

Level 1 (pertinent to O-288)

Bruce A and Bruce B operation is conducted through a full set of operating procedures in place covering normal and abnormal conditions that comply with the relevant provisions in the PROL and accompanying LCH [5] [6] including those in CSA N286-05 relating to the operating procedures.

BP-OPP-00002 Operating Policies and Principles - Bruce A and BP-OPP-00001 Operating Policies and Principles - Bruce B, identify the policies and principles agreed with the CNSC that drive the programs and processes to comply with these aforementioned requirements. The OPP, structured the same way in Bruce A and Bruce B, is subdivided into general and specific subjects. The front end covers multiple SSCs, records, reporting, while from Section 21 onwards it covers requirements for specific SSCs. Duties of responsible individuals during normal and abnormal operation are defined in Section 01.

BP-PROG-12.01 Conduct of Plant Operations covers Operations Documentation and Plant Operation for normal and abnormal operation. Operations Documentation procedures include Operating Manuals, Operating Memos, Alarm Response, Manuals, and Safety System tests. Procedures for the safe and reliable operation of plant equipment are prepared, approved, controlled and readily available to the operating staff. These procedures are prepared for anticipated normal, abnormal and emergency conditions. In addition to these procedures covering normal and abnormal plant states, Standard Operating Guidelines (SOGs) cover a set of other hazards, including bio-hazards, emergency vehicle response safety, fire pumpers, site emergency vehicles. These fall under the BP EST series from BP-EST-00101 to BP EST 02005, which are the responsibility of the Emergency Protective Services.

Operating procedures are created as controlled documents, in accordance with the requirements of BP-PROG-03.01 Document Management to ensure that document lifecycle management requirements of BP-PROC-00068, Controlled Document Life Cycle Management are met. Each Operating Procedure is reviewed, verified and validated before being approved and distributed for use. Procedures are continually verified and validated as part of Operator Training exercise and have been confirmed during Simulator Testing and as part of routine use, Maintenance, Testing and Operating personnel stop to ask questions if a procedure is unclear or cannot be executed as written and Station Condition Records are raised against procedures found to need improvement.

Bruce Power provisions for Level 1 for this safety principle are effective and adequate.

Levels 2 – 4 (pertinent to O-290)

Bruce A and Bruce B operation is conducted through a full set of operating procedures in place covering normal and abnormal conditions that comply with the relevant provisions in the PROL and accompanying LCH [5] [6] including those in CSA N286-05 relating to the operating procedures.

BP-OPP-00002 Operating Policies and Principles - Bruce A and BP-OPP-00001 Operating Policies and Principles - Bruce B, identify the policies and principles agreed with the CNSC that drive the programs and processes to comply with these aforementioned requirements. The OPP, structured the same way in Bruce A and Bruce B, is subdivided into general and specific subjects. The front end covers multiple SSCs, records, reporting, while from Section 21 onwards it covers requirements for specific SSCs. Duties of responsible individuals during normal and abnormal operation are defined in Section 01.

BP-PROG-12.01 Conduct of Plant Operations covers Operations Documentation and Plant Operation for normal and abnormal operation. Operations Documentation procedures include Operating Manuals, Operating Memos, Alarm Response, Manuals, and Safety System tests. Procedures for the safe and reliable operation of plant equipment are prepared, approved, controlled and readily available to the operating staff. These procedures are prepared for anticipated normal, abnormal and emergency conditions. In addition to these procedures covering normal and abnormal plant states, SOGs cover a set of other hazards, including bio-hazards, emergency vehicle response safety, fire pumpers, site emergency vehicles. These fall under the BP-EST-series from BP-EST-00101 to BP-EST-02005, which are the responsibility of the Emergency Protective Services.

BP-PROG-08.01 Emergency Measures Program describes how risks that have the potential to impact reactor safety, public safety, employee and responder safety, environmental safety and corporate reputation are managed through a risk-based program of prevention, mitigation, preparedness, response, and recovery. The program identifies several procedures and plans detailing different aspects of operations in emergency situations which can be classified as anticipated operational occurrences, design basis accidents or design extension condition depending on the extent of condition. The operation-based implementing documents are the Abnormal Incidents Manuals (AIMs) which include procedures that are specifically established to mitigate various design basis events, and SAMG for use if the plant has entered, or is going to enter, a state outside its design and analysis base.

A comprehensive set of Bruce Power specific AIMs and SAMG procedures are in place. The technical basis, entry and exit conditions, and assumptions used in AIM procedures make use of the deterministic analysis of the design basis events, while those used in SAMG technical basis are largely based on the deterministic safety analysis of severe BDBAs analyzed within PSA Level 2 scope, as well as PSA Level 1 and 2.

Operating procedures are created as controlled documents, in accordance with the requirements of BP-PROG-03.01 Document Management to ensure that document lifecycle management requirements of BP-PROC-00068, Controlled Document Life Cycle Management are met. Each Operating Procedure is reviewed, verified and validated before being approved and distributed for use. Procedures are continually verified and validated as part of Operator Training exercise and have been confirmed during Simulator Testing and as part of routine use, Maintenance, Testing and Operating personnel stop to ask questions if a procedure is unclear or cannot be executed as written and Station Condition Records are raised against procedures found to need improvement.

The Worker Learning and Qualification program (BP-PROG-02.02) satisfies the worker qualification and worker training requirements of applicable Bruce Power Licences and governing acts, codes and standards as referenced in BP-MSM-1 Sheet 0003, MSM - List of Applicable Governing Acts, Codes & Standards - Sheet 0003. As stated in BP-PROG-02.02, training programs based on the work performed by personnel are systematically developed and implemented so that the required competency is achieved and maintained. The procedures and job aids required to implement the Worker Learning and Qualification program allow the training elements that support Worker Qualifications, to be created, managed, and conducted using a Systematic Approach to Training (SAT). The Bruce Power SAT methodology satisfies the requirements for an iterative and interactive approach to the design of training.

Bruce Power provisions for Levels 2-4 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to O-288 Normal operating procedures and O-290 Emergency operating procedures

There were no strengths identified in SFRs from the standpoint of normal operating procedures and emergency operating procedures.

There is 1 GIO that will further improve normal operating procedures and emergency operating procedures.

GIO No.	GIO TITLE
GIO-043	Validation of Human Credited Actions

Summary

There are effective and adequate provisions in place for defence-in-depth Levels 1, 2, 3 and 4, which are the levels applicable to these safety principles. One GIO was identified that will further improve the provisions for these safety principles.

18.3.9. Safety Principles Related to Defence-in-Depth Level 3

There are 5 Safety principles related to DID Level 3. Bruce A and Bruce B design and operation is aligned with all the safety principles as demonstrated below.

D-168	Automatic safety systems
D-174	Reliability targets
D-177	Dependent failures
D-182	Equipment qualification
D-237	Control of accidents within the design basis

D-168 Automatic safety systems
Principle: Automatic systems are provided that would safely shut down the reactor, maintain it in a shut down and cooled state, and limit any release of fission products that might possibly ensue, if operating conditions were to exceed predetermined set points.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 21 for DID Level 3, as described below.

Level 3

At Bruce A and Bruce B there are four automatic special safety systems incorporated into the plant design to limit radioactive releases following an abnormal event. They are designed to mitigate the consequences of both a single failure in a process system and a (much less frequent) dual failure, consisting of a single failure in a process system combined with the coincident unavailability of one of the special safety systems. The four special safety systems are:

- Shutdown System 1 (SDS1);
- Shutdown System 2 (SDS2);
- Negative Pressure Containment (NPC) system;
- Emergency Coolant Injection (ECI) system.

To effectively reduce the risk presented by a postulated process system failure, special safety systems are independent of process systems, including the reactor regulating system, whose failure might require the subsequent action of the special safety system.

The two shutdown systems, SDS1 and SDS2, are functionally and physically independent of each other and functionally independent of the reactor regulating system that is achieved by employing diverse shutdown principles. SDS1 uses solid shutoff rods actuated electro-mechanically and driven by gravity. SDS2 actuates electro-mechanically and directly injects poison into the moderator hydraulically.

Redundant components are used where possible, so that the failure of a single component does not cause system failure. This leads to the use of 2 out of 3 voting logic, or channels, in many standby systems, which requires 2 of 3 separate instruments to fail before the system logic fails.

This type of logic also permits on-power testing, channel by channel, without impairing the functionality of the system, and prevents spurious initiation of a system if one instrument or channel fails.

In addition the following systems provide safety related functions to maintain the plant in a shut down and cooled state, and limit any release of fission products

- Emergency Boiler Cooling System (Bruce A) and Emergency Water System (Bruce B) provide feedwater to steam generators to ensure adequate decay heat removal in the event of a main steam line break resulting in the loss of normal feed.
- Qualified Power System (Bruce A) and Emergency Power Supply System (Bruce B) provide power for the equipment and instrumentation required to maintain and monitor the reactors in a safe shutdown state following a set of events leading to total loss of normal and backup power supplies.
- Powerhouse Emergency Venting System is a standby safety support system designed to mitigate the consequences following a steam piping or feedwater piping failure.
- Secondary Control Areas (SCA) are provided for post accident monitoring and to execute basic safety functions following any incident that renders the main control room uninhabitable due to fire, smoke, or excessive radiation fields.

These systems provide reactor shutdown, maintain the reactor in a shut down and cooled state, and limit any release of fission products that might possibly ensue should parameters exceed predetermined setpoints. Bruce Power provisions for Level 3 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-168 Automatic safety systems

There were no strengths identified in SFRs from the standpoint of automatic safety systems.

There are 7 GIOs that will further improve automatic safety systems.

GIO No.	GIO TITLE
GIO-026	BA & BB New Neutronic Trips
GIO-028	Upgrade Emergency and Standby Power Supplies
GIO-034	Safety System Reliability
GIO-036	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
GIO-070	Air Operated Valves-Replacement
GIO-076	DCC Cables and WIBAs –Replacement

GIO No.	GIO TITLE
GIO-090	SDS2 Enhancements

Summary

There are effective and adequate provisions in place for defence-in-depth Level 3, which is the level applicable to this safety principle. Seven GIOs were identified that will further improve the provisions for this safety principle.

D-174 Reliability targets

Principle: Reliability targets are assigned to safety systems or functions. The targets are established on the basis of the safety objectives and are consistent with the roles of the systems or functions in different accident sequences. Provision is made for testing and inspection of components and systems for which reliability targets have been set.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 22 for DID Level 3, as described below.

Level 3

The reliability targets are established on the basis of the safety objectives and are consistent with the roles of the systems or functions in different accident sequences To provide a high degree of assurance that a special safety system will perform as designed when called upon to do so, the unavailability target of each is limited to less than 10^{-3} yr/yr. Also, where such choice is available, special safety system components are designed such that the most likely failure modes are in the failsafe direction.

BP-PROG-11.01 Equipment Reliability and BP-PROG-11.04 Plant Maintenance and their implementing procedures describe the requirements and guidance for testing, inspection and maintenance of components and systems for which reliability targets have been set. The implementing procedures deal with scoping and identification of critical SSCs, continuing equipment reliability improvement, preventive maintenance implementation, performance monitoring, equipment reliability problem identification and resolution, long-term planning and life-cycle management.

Bruce Power provisions for Level 3 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-174 Reliability targets

There were no strengths identified in SFRs from the standpoint of improve reliability targets.

There are 5 GIOs that will further improve reliability targets.

GIO No.	GIO TITLE
GIO-028	Upgrade Emergency and Standby Power Supplies
GIO-034	Safety System Reliability
GIO-070	Air Operated Valves-Replacement
GIO-090	SDS2 Enhancements
GIO-095	45VDC Power Supplies-Replacement

Summary

There are effective and adequate provisions in place for defence-in-depth Level 3, which is the level applicable to this safety principle. Five GIOs were identified that will further improve the provisions for this safety principle.

D-177 Dependent failures

Principle: Design provisions seek to prevent the loss of safety functions due to damage to several components, systems or structures resulting from a common cause.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 23 for DID Level 3, as described below.

Level 3

The Bruce A and Bruce B design includes protection against common mode events including:

- Seismic Qualification

  At Bruce A, the seismic event for which this assurance is sought is called the Review Level Earthquake (RLE), and the shutdown period at least 72 hours. To address Seismic Qualification for Bruce A, a Seismic Margin Assessment (SMA) was conducted. The assessment was in accordance with the SMA guidelines of EPRI NP-6041SL, with modifications to fit the unique characteristics of the CANDU reactor system. The SMA is based on the evaluation of all structures, systems and components that make up the "success path", including the reactors and their auxiliary systems, control systems, electrical systems, as well as the civil structures. The electrical and mechanical components are captured on the Safe Shutdown Equipment List (SSEL).

  At Bruce B, the seismic design approach is different than for Bruce A. Seismic design philosophy is presented in Section 2.5.2 of Part 2 of the Bruce B Safety Report. Dynamic analyses of structures were done based on both lumped mass and finite element models to determine the predominant frequencies and modal displacements of the structures. The seismic response of the structures was determined by modal analysis using both the artificial time history and the response spectra method as seismic input. An artificial time history motion, whose response spectra curve envelopes the design ground response spectra curves, was developed. This time history ground motion was applied as input in the seismic analyses of the nuclear structures to produce acceleration floor response spectra. These floor response spectra were applied as input for the seismic qualification of seismically qualified equipment and systems. Details of seismic analysis and qualification of the reactor assembly are provided in Section 2.5.2.3 of Part 2 of the Safety Report. Specific safety related systems that are necessary for the orderly shutdown of the reactor, for the maintenance of the reactor in the safe shutdown state for an indefinite period, and for the removal of decay heat from the fuel for an indefinite period, have been designed and constructed to withstand the specified earthquake. In addition, non-qualified systems whose failure could cause the failure of qualified systems have been seismically restrained. The list of systems and structures specified as seismically qualified and their level of qualification is provided in Seismic Qualification of Safety-Related Systems Design Guide, NK29-DG-03650-002. The Design Guide also specifies the basic design approach. Since the HT system is designed to withstand a DBE, a DBE will not cause a loss of coolant accident. A loss of coolant accident coincident with a DBE is not part of the design basis.

A Probabilistic Seismic Hazard Assessment was done for the Bruce B site in 2011 [NK29-03500.8 P NSAS, Rev.1] which provides information about earthquakes beyond the DBE level.

- Missile Protection

  Features incorporated into the Bruce A and Bruce B designs provide an adequate level of protection against any credible turbine generator missile. These features include:

  - Separation of the 600 V Class II switchgear, such that a single missile cannot disable both halves of the system.
  - Reinforced concrete barriers, such that a turbine generator missile cannot strike the HT pump motors.
  - Adoption of separation measures, such that a single missile cannot disable sufficient equipment to prevent safe shutdown, monitoring, or decay heat removal.

- Protection Against Dynamic Effects Associated with the Rupture of Piping

  The Bruce A and Bruce B Heat Transport system contains large components, connected into the system, which penetrate the primary containment boundary. These components are all supported and restrained in such a way that the containment envelope will not be damaged as a result of the thrust forces caused by any credible failure of piping connected to the component.

  It has been shown by accident analyses that, after postulated pipe failures, the reactor would be safely shut down, decay heat removal capability would be available and adequate containment integrity would be maintained.

  The original SSC design did not consider pipe whip and jet impingement. As part of Units 1 and 2 restart, a project was initiated for Bruce A with three phases (develop the methodology; assess piping inside the reactor vault; and, assess piping outside the reactor vault). As discussed in [36], all three phases have been completed for Bruce A. The assessments concluded that protection of equipment against pipe whip and jet impingement for high-energy piping inside the Unit 2 reactor vault was in compliance with the practices, expectations, and guidelines in modern standards such as IAEA NS-G-1.11. The relevant differences between the layout of high-energy piping larger than NPS 6 inside the reactor vaults of Bruce Units 1, 3 and 4 were then assessed against those of Unit 2 and the conclusions from the pipe whip and jet impingement assessment of Unit 2 are also valid for the other Bruce A units. A similar assessment of protection of equipment against pipe whip and jet impingement for high-energy piping inside of the Bruce A Unit 0 demonstrated that the layout of the piping is in compliance with the practices, expectations, and guidelines in modern standards such as IAEA NS-G-1.11.

A similar approach will be executed for Bruce B.

- Fire Protection

  The original design of Bruce A and B did not consider the potential for fires and explosions, although the effects of such events were addressed, and features were provided to protect against them. To address this gap, a Fire PSA has been prepared for both Bruce A and B as part of an on-going project to implement the CNSC Regulatory Standard S-294 in support of the operating licence renewal in 2014 [NK21-CORR-00531-11005 / NK29-CORR-00531-11397]. Bruce Power has also completed significant operational and design improvements to align with modern codes and standards on fire protection. Some of these improvements are in progress and are included in the IIP as listed in Table 31 in Section 9.3.

- Environmental Qualification of Safety-Related Equipment

  Bruce A and Bruce B Environmental Qualification (EQ) provides the documented assurance that essential safety-related systems, components and structures are capable of performing their functions when subjected to the environmentally harsh conditions that could result from postulated DBAs.

  Essential systems, components, and structures provide a safety function in accordance with the design and licensing basis of the station and consistent with the assumptions and requirements documented in current accident analysis documented in Part 3 of Bruce A and Bruce B Safety Reports. All design basis accidents (single and dual failure), with the potential to cause common mode equipment failures are considered. For each such accident, a reliable and qualified line of defence is provided to achieve the basic nuclear safety functions, i.e., achieve and maintain reactor shutdown (Control), remove fuel heat (Cool), contain radioactive contamination (Contain) and monitor post-accident conditions (Monitor).

  The determination of environmental conditions considers the spectrum of break sizes, location of credible breaks, credible consequential failures that are the result of the DBA, and the possible continued operation of selected non-environmentally qualified systems.

  The environmentally harsh conditions that these essential safety-related systems may experience include exposure to radiation, temperature, pressure, humidity, and chemical effects. The conditions will vary depending on the location of the equipment in the plant, and on the nature of the postulated DBA. These conditions have been thoroughly evaluated for all DBA categories considered and have been documented in the Room Conditions Manual.

- Bruce A and Bruce B design also include provisions to prevent the loss of safety functions due to damage to several components, systems or structures resulting from a common cause:
  - The two shutdown systems, SDS1 and SDS2, are functionally and physically independent of each other and functionally independent of the reactor regulating system.
  - Independence is achieved by employing diverse shutdown principles, i.e., SDS1 uses solid shutoff rods (gravity driven), and SDS2 directly injects liquid poison into the moderator (pressurized injection).
  - The systems are also geographically separated. The shutoff rods are inserted vertically into the top of the reactor. The poison injection tubes are inserted horizontally into the side of the reactor.
  - Ancillary mechanical and process equipment is similarly separated. The shutoff rod drives are located above the reactor, whereas the poison supply system is located to the side of the reactor. The measurement elements for the two systems are physically separated as well.

  Separation of the instrumentation channels of the two systems is achieved by channelization. Each of the three channels on a specific special safety system follows a separate route. This does not exclude that one of the triplicated channels on one special safety system may follow a common route with one of the associated triplicated channels of another special safety system, i.e., associated channels. Adequate separation is maintained with three different routes for three sets of associated channels. Channelization ensures that the three cable routes are separated, that the equipment associated with the three sets of channels is located in three different rooms, and that power to the three sets of channels is supplied by three different buses. Consequently, any credible local common mode event can affect only one set of channels, leaving the other two unimpaired and thus the special safety systems remain functional.

  Each safety system's initiation logic is independent from each other and from process systems. SDS1 uses general coincidence logic, whereas SDS2 uses local coincidence logic to increase diversity.

  Bruce Power undertook, as part of its disposition of Fukushima Action Items, a re-evaluation of the site-specific magnitudes of each external event to which the plant might be susceptible, using modern calculations and methods; and an evaluation as to whether the current site-specific design protection for each external event so assessed is sufficient. An extensive screening assessment was conducted based on a screening methodology submitted to CNSC. The list covers all the external hazards, as well as several hazards that could be classified as internal hazards. These hazards were initially subjected to a first-level screening, and the hazards which were not eliminated in the first level were then subjected to a second level of screening which resulted in the only hazards requiring assessment were tornadoes, high winds and external flooding. To address these remaining external hazards, Bruce Power has developed a methodology for analysis of tornadoes, high winds and external flooding and submitted the following reports to the CNSC:

- a High Wind PRA Report (which includes tornado hazard assessment), Seismic PRA Report, Fire PRA Report; and
- External Flooding Assessment (in addition to revised versions of a Seismic PRA Report and Fire PRA Report).

The effect of ageing on the plant capability to withstand internal and external hazards is managed by:

- an ageing management program that includes equipment lifecycle management and fitness-for-service evaluations; and
- a DSA/PSA update process to incorporate up-to-date plant-specific component condition and performance data.

The above demonstrates that safety functions will not be lost due to a common-cause event. Bruce Power provisions for Level 3 for this safety principle are effective and adequate.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-177 Dependent failures

There were no strengths identified in SFRs from the standpoint of dependent failures.

There are 4 GIOs that will further improve dependent failures.

GIO No.	GIO TITLE
GIO-003	Assess pipe whip and jet impingement
GIO-019	Assess and improve seismic Qualification
GIO-091	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
GIO-092	Bruce B Fire Protection Upgrades to Align with CSA-N293-07

Summary

There are effective and adequate provisions in place for defence-in-depth Level 3, which is the level applicable to this safety principle. Four GIOs were identified that will further improve the provisions for this safety principle.

D-182 Equipment qualification
Principle: Safety components and systems are chosen that are qualified for the environmental conditions that would prevail if they were required to function. The effects of ageing on normal and abnormal functioning are considered in design and qualification.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 24 for DID Level 3, as described below.

Level 3

Environmental Qualification (EQ) was not a requirement in the original design of Bruce A and Bruce B. EQ was implemented as an improvement project. Programmatic requirements were incorporated in Bruce Power governance for the qualification of equipment important to safety to ensure that they are capable of fulfilling their safety functions as installed and for maintenance of EQ for the operating life of the plant.

Processes and procedural barriers that are included in Engineering Change Control (ECC) and EQ flags/instructions are available in PassPort (Bruce Power's data/information management system), as well as maintenance predefineds and EQ Bills of Materials. As required, engineering, operations and maintenance staff are trained on EQ considerations and requirements. These measures will effectively maintain EQ for the life of the plant, both for conditions that occur during normal operation and those that occur for Design Basis Accidents, and for less frequent internal and external events, such as seismic events.

The scope for the environmental qualification program includes all components, which are essential to provide a safety function consistent with the assumptions and requirements documented in the accident analysis, in accordance with the design and licensing basis established for each station.

Among the main systems (other than the systems and components that make up the pressure boundary) subject to environmental qualification are all or parts of the following:

- Emergency Coolant Injection (ECI).
- Shutdown System 1 (SDS1).
- Shutdown System 2 (SDS2).
- Negative Pressure Containment (NPC) System.
- Qualified Power Supply (QPS) System (Bruce A).
- Emergency Power Supply (Bruce B)
- Emergency Boiler Cooling System (EBCS) (Bruce A).
- Emergency Water System (Bruce B).
- Moderator and Moderator Auxiliary Systems.
- Heat Transport (HT) and HT Auxiliary Systems.

- Powerhouse Emergency Venting System (PEVS).
- Fuel Handling and Irradiated Fuel Bay Systems.

The EQ process consists of the following general steps:

- Identifying the Design Basis Accidents that result in post-accident Harsh Environments (HE) with the potential to cause common mode failures.
- Defining the Room Conditions. This involves the determination of the normal, accident, and post-accident conditions that qualified equipment is expected to experience and tolerate (for the duration of the mission time) under various DBAs as a function of location in the plant.
- Defining the EQ nuclear safety requirements in a Safety Requirement Matrix (SRM) for components exposed to harsh environment. Components or structures determined to be subjected to only normal design conditions or located in areas of mild conditions are exempt from further assessment.
- Conducting a failure mode and effect analysis. If a component has an adverse failure mode in a harsh environment, it is placed on the EQ List and a Component EQ Verification and Requirements sheet is prepared and included in an EQ Dossier. The Dossier is a plant specific document. It is a design assurance document that documents the capability of EQ equipment and components to perform safety-related functions under the environmental stress of the applicable design basis accidents.
- EQ includes the effects of ageing during normal plant operation in the environmental qualification process (i.e., EQ Assessments), so the procedures used to monitor normal plant conditions are identified, as these are important to ensure that the equipment qualification is maintained for the life of the plant. To maintain environmentally qualified equipment in its qualified state, after EQ has been initially established, procedural barriers are in place within ECC and EQ flags/instructions are available in PassPort including maintenance predefineds and EQ Bills of Material. Engineering, operations and maintenance staff are trained as required on EQ considerations.

The above demonstrates that there is an EQ program in place that will maintain environmental qualification over the life of the station. The program considers the effects of aging on SSC performance and qualification. Bruce Power provisions for Level 3 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-182 Equipment qualification

SF-03-S1	The quality of the programmatic documents (i.e., programs and procedures) for the equipment qualification process is very good, with interfaces with other station procedures well identified, recent revisions and updating for most procedures, and incorporation of issues identified in audits and self-assessments.

SF-03-S2	The IAEA OSART review of Bruce B completed in 2015 reviewed all aspects of the environmental qualification program and recognized its overall implementation as "good performance". Therefore, the management of the EQ program is considered to be a strength in this report.

There are 3 GIOs that will further improve equipment qualification.

GIO No.	GIO TITLE
GIO-003	Assess pipe whip and jet impingement
GIO-019	Assess and improve seismic margins
GIO-036	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment

Summary

There are effective and adequate provisions in place for defence-in-depth Level 3, which is the level applicable to this safety principle. Three GIOs were identified that will further improve the provisions for this safety principle.

D-237 Control of accidents within the design basis

Principle: Provisions are made at the design stage for the control of accidents within the design basis, including the specification of information and instrumentation needed by the plant staff for following and intervening in the course of accidents.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 51 for DID Level 3, as described below.

Level 3

At Bruce A and Bruce B there are four special safety systems incorporated into the plant design to limit radioactive releases following an abnormal event. They are designed to mitigate the consequences of both a single failure in a process system and a (much less frequent) dual failure, consisting of a single failure in a process system combined with the coincident unavailability of one of the special safety systems. The four special safety systems are:

- Shutdown System 1 (SDS1).
- Shutdown System 2 (SDS2).
- Negative Pressure Containment (NPC) system.
- Emergency Coolant Injection (ECI) system.

These systems are controlled either automatically, whenever certain parameters exceed specified bounds or manually (in accordance with the operating documentation) by the use of trip buttons in the control room and also as applicable in the Secondary Control Areas (SCAs).

These systems are independent of each other. They also are, as much as possible, independent of any of the process systems, including the reactor regulating system.

To provide a high degree of assurance that a special safety system will perform as designed when called upon to do so, the unavailability target of each is limited to less than 10^{-3} yr/yr. Also, where such choice is available, special safety system components are designed such that the most likely failure modes are in the failsafe direction.

Each process and nuclear measurement loop that is essential for the operation of a special safety system is redundantly designed, usually triplicated, such that a single loop component or power supply failure will not incapacitate or spuriously invoke operation of the special safety system.

Safety System Monitoring Computer (SSMC) is a computer system used to monitor the state of the shutdown and ECI systems. For each unit the system consists of a monitoring computer optically linked to nine intelligent multiplexers, one for each channel of the two shutdown systems, and one for each channel of the emergency coolant injection system. In addition, a station safety system monitoring computer, optically linked to three intelligent multiplexers, is used to monitor the common portions of the emergency coolant injection system.

The containment system, excluding the Emergency Filtered Air Discharge System (EFADS), is designed to be self-actuating.

Consequently, few control systems directly related to the containment function are required. The control functions that are provided are as follows:

- Power operated auxiliary pressure relief valves around the main pressure relief valves maintain the containment at a slightly negative pressure following an accident. These valves have a manual override capability.
- The four instrumented pressure relief valves are self-actuating, but their control system is initiated at a 20% valve lift. They can also be manually controlled.
- Automatic containment isolation occurs on high containment pressure or high containment activity. These signals activate closure of the containment isolation dampers.
- The emergency filtered air discharge system components are controlled during long-term post-LOCA operation.

Two independent channels, N and P, are provided for instrumentation and control logic.

SSCs to shut down, control, cool, and monitor have been included in the design for DBA and BDBA conditions. Bruce Power provisions for Level 3 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to D-237 Control of accidents within the design basis

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.

There are 11 GIOs included in the IIP that will further improve control of accidents within the design basis.

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response
GIO-026	BA & BB New Neutronic Trips
GIO-028	Upgrade Emergency and Standby Power Supplies
GIO-034	Safety System Reliability
GIO-036	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
GIO-070	Air Operated Valves-Replacement
GIO-076	DCC Cables and WIBAs –Replacement
GIO-090	SDS2 Enhancements
GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
GIO-101	M/34720 Relief Valves For Overpressure Protection
GIO-102	I/63472 Remote Relief Valve Position Indication

Summary

There are effective and adequate provisions in place for defence-in-depth Level 3, which is the level applicable to this safety principle. Eleven GIOs were identified that will further improve the provisions for this safety principle.

18.3.10. Safety Principles Related to Defence-in-Depth Level 4

There are 3 Safety principles related to DID Level 4. Bruce A and Bruce B design and operation is aligned with all the safety principles as demonstrated below.

AM-318	Strategy for accident management
AM-323	Training and procedures for accident management
AM-326	Engineered features for accident management

AM-318 Strategy for accident management
Principle: The results of an analysis of the response of the plant to potential accidents beyond the design basis are used in preparing guidance on an accident management strategy.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 74 for DID Level 4, as described below.

Level 4

BP-PROG-08.01 Emergency Measures Program is developed to enable effective response to all hazards at Bruce Power by considering:

- Design basis accidents;
- Beyond design basis accidents;
- Other emergencies (e.g., conventional) leading to nuclear emergencies; and
- Multi-unit accident scenarios, if applicable.

The relevant operation-based implementing documents are the AIMs (Abnormal Incident Manual), which include procedures specifically established to mitigate various design basis events. SAMG is used if the plant has entered, or is going to enter, a severe accident condition. A comprehensive set of Bruce Power specific AIMs and SAMG documents describe the technical basis, entry and exit conditions, and assumptions used in AIM procedures, including credits for operator actions that make use of the deterministic analysis of the design basis events. SAMG technical bases are largely based on the deterministic safety analysis of severe BDBAs analyzed within the PSA Level 2 scope, as well as previous PSA Level 1 and 2 assessments.

As part of the current SAM program, Bruce Power has issued a number of SAMG documents, including a hierarchy of guides and procedures implementing the SAM procedure, under the Technical Support Group User’s Guide. The hierarchy defines conditions for entry into a SAM process, and it contains a structured set of SAM tools (e.g., a Diagnostic Flow Chart, personnel instructions and a severe challenge status tree) to provide a pre-planned, systematic approach to guide the plant response in case of a severe accident.

SAMG has been updated to implement improvements proposed in the COG joint project JP4426 in response to the events at the Fukushima Daiichi plant. The scope of the project included responses for multi-unit and IFB events in severe accident conditions, and SAMG for shut down units or low-power operation.

Bruce Power provisions for Level 4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to AM-318 Strategy for accident management

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.
SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There are 2 GIOs that will further improve strategy for accident management.

GIO No.	GIO TITLE
GIO-011	Implement enhancements to SAMG
GIO-089	Whole-Site Probabilistic Risk Assessment

Summary

There are effective and adequate provisions in place for defence-in-depth Level 4, which is the level applicable to this safety principle. Two GIOs were identified that will further improve the provisions for this safety principle.

AM-323 Training and procedures for accident management

Principle: Nuclear plant staff are trained and retrained in the procedures to follow if an accident occurs that exceeds the design basis of the plant.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 75 for DID Level 4, as described below.

Level 4

TQD-00005 Emergency Response Organization Training and Qualification Description establishes the requirements for the training and qualification of individuals assigned to specific emergency response positions as defined in BP-PLAN-00001 Nuclear Emergency Response Plan, following a systematic approach to training methodology. BP-PROC-00010 Emergency Preparedness Drills and Exercises provides a comprehensive list of drill and exercise objectives and provides for a schedule for conducting drills and exercises such that all of the objectives are tested within a set period of time. The schedule is reviewed at least quarterly. The CNSC is included on the distribution list.

Bruce Power provisions for Level 4 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to AM-323 Training and procedures for accident management

SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There are no additional planned initiatives included in the IIP that will further improve training and procedures for accident management.

Summary

There are effective and adequate provisions in place for defence-in-depth Level 4, which is the level applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

AM-326 Engineered features for accident management

Principle: Equipment, instrumentation and diagnostic aids are available to operators, who may at some time be faced with the need to control the course and consequences of an accident beyond the design basis.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 76 for DID Level 4, as described below.

Level 4

Bruce A and Bruce B have a number of complementary design features for management of BDBAs.

For mitigation of hydrogen following the low probability design basis event, combinations of two separate systems are provided.

- Hydrogen Ignition System for mitigation of short term hydrogen generation; and
- Passive Autocatalytic Recombiners (PARs) for slower longer term hydrogen generation such as from radiolysis of water. PARs provide defence-in-depth for short term hydrogen mitigation as well.

EFADS is operated to control long-term radiological dose to the public and station staff by providing a well defined, filtered, controlled and monitored release path of fission products from containment following a LOCA or other Design Basis Accidents. Several upgrades to the system are underway to meet the reliability target and the current Provincial Emergency Preparedness guidelines.

As part of the Fukushima Follow-up Bruce Power recognized the need to address multi-unit events including a station blackout. Update of SAMG and its implementation to address multi-unit events is ongoing. For example, options for enhancing the ability of containment to accommodate severe accidents in multiple units follow:

- The ongoing analysis involves numerous multi-unit event combinations with various credits for mitigating actions and systems. The analysis includes an evaluation of the benefits and practicality of installing passive filtered venting.
- Bruce Power has installed containment bypass tees and containment boundary valves into the existing EFADS piping where it exits the Vacuum Building and Pressure Relief Valve (PRV) manifold at Bruce A and B. The purpose of the bypass line and isolation valves is to allow a containment filtered venting system to be installed at a later date without the need for an additional containment outage. An assessment of options for ensuring containment integrity and filtered venting in the event of a multi-unit severe accident has concluded that existing design capability and emergency mitigation measures are a viable alternative to the installation of a filter vent system. Furthermore, Bruce Power performed supplementary evaluations of improvements that strengthen defence-in-depth, which showed that the best option to maintain containment integrity is a passive CFVS.
- Post-Fukushima design enhancements to prevent and mitigate severe accidents that are in progress include adding design features to allow external water makeup to the HTS, moderator system, steam generators and the irradiated fuel bay, as well as enhancements to the emergency power supply and providing overpressure protection to the shield tank. These modifications are intended to provide further defence-in-depth against beyond design basis accidents and to support SAMGs by early mitigation of the severe accident progression and protecting containment integrity. These modifications significantly improve the fourth level of defence-in-depth. PSAs, taking into account Emergency Mitigation Equipment, demonstrate significant improvements in Severe Core Damage Frequency (SCDF) and releases.

The Post-Accident Radiation Monitoring System (PARMS) provides on-line radioisotopic analysis for noble gases, gross gamma detection and off-line radioisotopic analyses for particulates, iodine and tritium. Several upgrades are underway to meet the performance requirement in terms of providing data for all single or dual failure accidents. The PARMS instrumentation and equipment will cope with a wide range of accident scenarios including many BDBAs and severe accidents.

The control computers and the SSMC can record and display the parameters that are important to safety. This information will be used to monitor the course of DBAs and provide information on the status of essential equipment. All of the necessary instrumentation for monitoring essential information is available in the main control room (and SCA) and are seismically qualified. Should the DCCs/SSMC (which are not seismically qualified) not be available there would be a need to rely on manual record keeping for trends.

Equipment, instrumentation, and diagnostic aids (SAMG documentation) are in place. Bruce Power provisions for Level 4 for this safety principle are effective and adequate. This is further corroborated by the strengths listed below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to AM-326 Engineered features for accident management

SF-05-S2	Bruce Power has implemented or is in the process of adding significant preventive and mitigating design modifications that are intended to provide further defence-in-depth against design basis events and severe accidents and to support SAMGs by mitigating severe accident progression and protecting containment integrity.
SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There is 1 GIO that will further improve engineered features for accident management.

GIO No.	GIO TITLE
GIO-002	Implement design changes to improve severe accident response

Summary

There are effective and adequate provisions in place for defence-in-depth Level 4, which is the level applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

18.3.11. Safety Principles Related to Defence-in-Depth Level 5

There are 2 Safety principles related to DID Level 5. Bruce A and Bruce B design and operation is aligned with all the safety principles as demonstrated below.

S-140	Feasibility of emergency plans
EP-339	Assessment of accident consequences and radiological monitoring

S-140 Feasibility of emergency plans
Principle: The site selected for a nuclear power plant is compatible with the offsite countermeasures that may be necessary to limit the effects of accidental releases of radioactive substances, and is expected to remain compatible with such measures.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 78 for DID Level 5, as described below.

Level 5

The original plant did not explicitly consider feasibility of emergency plans as part of site selection criteria. BP-PROG-08.01 Emergency Measures Program and BP-PLAN-00001 Bruce Power Nuclear Emergency response Plan ensure compatibility with the offsite countermeasures that are necessary to limit the effects of accidental releases of radioactive substances on a continuing basis.

The on-site technical and operations support centres, i.e., the Main Control room, and the Emergency Operations Centre (EOC), are equipped with the necessary communications and other equipment as described in Section 4.1.2.2 of BP-PLAN-00001 Bruce Power Nuclear Emergency Response Plan. The emergency response plan maintenance requirements are defined in Section 4.1.3 of BP-PLAN-00001 Bruce Power Nuclear Emergency Response Plan. These include a variety of review and assessment mechanisms as further defined by implementing procedures, including maintenance and testing of equipment and facilities which include the EOC, drills and exercise, administrative requirements management, and program assessment (which includes quality assurance assessments, self-assessments, and independent assessments). These provide assurance of the process for ensuring the adequacy of these on-site centres. As indicated in Section 5.3.1 of BP-PLAN-00001 Bruce Power Nuclear Emergency Response Plan, facilities and equipment are being maintained on a routine basis.

In addition, Bruce Power's operating organization has given adequate consideration to significant changes at the site of the nuclear power plant and in its use, organizational changes at the plant, changes in the maintenance and storage of emergency equipment and developments around the site that could influence emergency planning. There have been no significant changes at the site such that consideration was required for changes to the emergency planning.

Bruce Power has recently consolidated the Site Management Centre and the Corporate Emergency Support Centre into an Emergency Management Centre (EMC) located at the Bruce Power Visitor's Centre in order to improve arrangements, including supporting the Incident Management System, and making ensuing changes to the emergency plan and procedures. The new EMC includes a back-up power supply to ensure the EMC is capable of providing continuous AC power to critical building loads and equipment for at least 72 hours after a BDBA. Emergency response crews have been trained and exercised multiple times in the deployment of the emergency equipment.

As a result of Fukushima Action Item completion, communications upgrades have been completed, including a radio communications infrastructure and satellite phone capability both at the new EMC and the Central Maintenance and Laundry Facility (CMLF). Further enhancements included the installation of a VSAT (Very Small Aperture Terminal) system at the EMC to provide multiple backup phone hubs and internet connectivity. These upgrades address connectivity issues between the EMC and station EOC, as well as external agencies.

As part of the emergency preparedness program, Bruce Power has submitted a transition plan to update its governance to achieve compliance with REGDOC-2.10.1 in accordance with the LCH.

On-site technical and operations emergency support centres are in place. Emergency response plans with supporting implementing procedures have been prepared and validated. An off-site Emergency Management Centre is available and equipped, including reliable networks for on-site and off-site communications. The implementation of the emergency measures is adequate to limit the effects of accidental releases of radioactive substances. Bruce Power provisions for Level 5 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to S-140 feasibility of emergency plans

SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There is 1 GIO included in the IIP that will further improve feasibility of emergency plans.

GIO No.	GIO TITLE
GIO-044	Emergency preparedness

Summary

There are effective and adequate provisions in place for defence-in-depth Level 5, which is the level applicable to this safety principle. One GIO was identified that will further improve the provisions for this safety principle.

EP-339 Assessment of accident consequences and radiological monitoring
Principle: Means are available to the responsible site staff to be used in early prediction of the extent and significance of any release of radioactive materials if an accident were to occur, for rapid and continuous assessment of the radiological situation, and for determining the need for protective measures.

Results of Review

This safety principle is mapped to SRS-46 Objective Tree Figure 78 for DID Level 5, as described below.

Level 5

As described in Section 4.2.2 of BP-PLAN-00001 Bruce Power Nuclear Emergency Response Plan, accident assessment techniques are employed to determine the extent of on-site radiation impact and to predict the off-site radiation consequence to the public. Processes and methodology include determination of system status from plant parameters, radiological source term measurements, core or fuel damage assessment techniques, meteorological information, off-site dose projections, source term sampling using post accident radiation monitoring systems, and dose assessment verification using off-site field survey results.

Appendix B and Appendix C of BP-PLAN-00001 Bruce Power Nuclear Emergency Response Plan provide the list of site-specific and generic documents that implement or support the use of accident assessment techniques. Dose projection estimates to the public living around the Bruce Power Site following an airborne release of radioactive materials are achieved using a software code called BERP (Bruce Emergency Response Projection) program. Bruce Power will run the BERP code in parallel with the Province. The BERP code results are used by the PEOC (Provincial Emergency Operations Centre) to aid with decision making regarding public protective actions such as sheltering or evacuation.

Processes are in place to enable personnel to determine station system status and, using supplementary information and the BERP code, predict public doses. The BERP code results are communicated to the PEOC. Continuous verification of dose assessments is performed using off-site field survey results. Bruce Power provisions for Level 5 for this safety principle are effective and adequate. This is further corroborated by the strength noted below.

Strengths Identified in SFRs and Global Improvement Opportunities in the IIP Relevant to EP-339 Assessment of accident consequences and radiological monitoring

SF-13-S1	As noted in sections 4 and 5.1, a particular strength was noted in emergency preparedness as a result of changes related, or in follow-up, to the lessons learned from the Fukushima events.

There are no GIOs that will further improve engineered features for assessment of accident consequences and radiological monitoring.

Summary

There are effective and adequate provisions in place for defence-in-depth Level 5, which is the level applicable to this safety principle. No GIOs were identified that would further improve the provisions for this safety principle.

18.4. Summary and Conclusions

The concept of defence-in-depth has evolved since it was originally applied to the design of early CANDU reactors. The approach used was to keep various levels of defence-in-depth independent of each other to the greatest extent practicable. For example, Level 1 defence-in-depth systems, i.e., process systems, are designed so that any failure in the system is not propagated to the control systems that control these processes. Similarly a failure in a control system does not propagate to the next level of defence-in-depth, i.e., the safety systems. This is accomplished through adequate separation of the control systems from the safety systems. Internationally this is achieved by ensuring adequate buffering of any components shared between the control and safety systems so that the failure cannot be propagated. In Canada, it has been done through complete separation of the control and safety systems. Level 2 defence-in-depth is achieved by measuring deviations from normal operating conditions by both the regulating system and the special safety systems. Digital computerized monitoring of parameters important to safety is used in the design of the reactor regulating system. Level 3 defence-in-depth includes the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. As part of defence-in-depth, pressure retaining components in any safety system are required to meet the highest design and quality standards. In summary, the original design incorporated adequate DID at Levels 1, 2 and 3 comparable to new nuclear power plants.

Level 4 defence-in-depth makes use of many systems that are not normally credited in Canadian safety analysis for design basis accidents. They are used to mitigate the consequences of a BDBA or a Severe Accident. Such accidents have a very low frequency and usually occur because safety systems have not been able to perform their function, either through multiple component failures within those systems or through loss of common services. They are generally backup process systems and as such would have been designed such that their failure would in no way affect the control or safety systems. Comprehensive on-site and off-site plans and new facilities and processes have been implemented for response to emergencies as the fifth level of defence. Significant improvements have been implemented in improving the fourth and fifth levels of defence based on new requirements of the CNSC and international OPEX since Bruce A and Bruce B were put into service.

The detailed review of the 53 Safety Principles described in this report has shown that Bruce A and Bruce B design and operation has appropriate provisions in all applicable levels of defence-in-depth and that significant improvements have been implemented since the plant was put into service. The review has also shown that the strengths identified during Safety Factor reviews, as well as relevant safety improvements identified in the IIP will further enhance defence-in-depth provisions at all levels in Bruce A and B design and operation. Additional improvement opportunities that have been included in the IIP based on the results of the PSR and GA described in this report will further enhance defence-in-depth provisions of Bruce A and

Bruce B. It should also be emphasized that the scope of improvement initiatives as part of on-going operations is much wider than those that have to be included in the IIP as part of the PSR process, as demonstrated in Table 31 in Section 9.3 of this report

A summary of the detailed review of the 53 Safety Principles, some of which are applicable to multiple levels of DID, is presented for each level of defence in the following sub-sections.

18.4.1. Level 1

The aim of the first level of defence is to prevent deviations from normal operation, and to prevent failures of structures, systems and components (SSCs) important to safety.

At Bruce A and Bruce B, Level 1 defence-in-depth includes conservative design and high-quality construction and commissioning which provides a baseline confidence that unexpected failure of SSCs and deviations from normal operations are minimized and accidents are prevented. Quality levels and engineering practices, such as the application of redundancy, independence, separation and diversity has been used in the designed commensurate with the safety importance of SSCs. Particular attention has been given in the original design for the provision of multiple barriers to protect the public and environment at large from radioactive hazards and to minimize risks associated with plant operation. Bruce A and Bruce B design provides the layers of defence against the release of fission products to the environment. These include:

- The uranium dioxide (UO_2) fuel, which contains almost all the radioactivity, is a ceramic with high melting point sealed in a corrosion resistant metallic cladding;
- The zirconium alloy fuel element sheath which has been demonstrated over forty years to have a very low failure rate;
- The Heat Transport system designed, manufactured, installed, tested and inspected to high quality requirements;
- The sub-atmospheric Containment System designed to retain a large fraction of any fission products released from the heat transport system following an accident;
- The Filtered Air Discharge System to remove particulates and iodine from controlled release following repressurization of containment;
- The exclusion boundary that provides a separation between the station and the public; and
- An emergency response centre and emergency response plans which are in place to mitigate the consequences of any release from the station.

Assessments have demonstrated that the plant has been designed conservatively (considering available OPEX at that time) using the appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication, plant construction and commissioning. Practicable design and operational improvements are implemented continuously based on national and international operating experience in addition to those driven by the evolving regulatory requirements.

The plant is operated within a prescribed safe operating envelope and conservatively by highly skilled and qualified staff. The condition of SSCs is well understood and plant safety and reliability is maintained through a set of systematic and planned surveillance, testing, inspection and maintenance activities using best industry practices and OPEX thus supporting prevention of unexpected failures and deviations from normal operation.

Bruce A and Bruce B design and operation aligns with the 35 Safety Principles related to Level 1 defence-in-depth. Improvements identified in 45 GIOs that are included in the IIP will further enhance Level 1 DID.

18.4.2. Level 2

The aim of the second level of defence is to detect and intercept deviations from normal operation, in order to prevent AOOs from escalating to accident conditions and to return the plant to a state of normal operation.

At Bruce A and Bruce B, Level 2 defence-in-depth is achieved by measuring deviations from normal operating conditions by both the regulating system and the special safety systems. Digital computerized monitoring of parameters important to safety is used in the design of the reactor regulating system. The process features of the regulating system (liquid zone control and setback function) and the safety features (stepback function) respond to deviations from normal operation before these deviations progress further necessitating the next level of defence to act. Reactor regulating system minimizes or excludes uncontrolled transients all but the most serious Postulated Initiating Events (PIEs). Furthermore, the Safety System Monitoring Computers are provided as an operator's aid to assist in the detection of abnormal conditions in the safety systems, which if left uncorrected might lead to impairment of the trip function or cause an unnecessary reactor trip.

SSCs important to plant safety and reliability are continuously monitored and tested to assure that they operate within their safe operating envelope and comply with associated reliability and performance requirements.

Bruce A and Bruce B design and operation aligns with the 32 Safety Principles related to Level 2 defence-in-depth. Improvements identified in 44 GIOs that are included in the IIP will further enhance Level 2 DID.

18.4.3. Level 3

The aim of the third level of defence is to minimize the consequences of accidents by providing inherent safety features, fail-safe design, additional equipment and mitigating procedures.

At Bruce A and Bruce B, Level 3 defence-in-depth includes the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions are capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features minimizes the need for operator actions in the early phase of a DBA.

Special safety systems are independent of process systems and RRS which effectively reduces the risk presented by a postulated process system failure.

To the greatest extent practicable, the special safety systems are also independent of each other in design and operation. This requirement evolves from the Canadian reactor safety principle of analyzing each postulated process system failure in conjunction with a failure of each of the special safety systems in turn.

As an additional feature, both SDS1 and SDS2 are capable of shutting the reactor down independently. The provision of two independent reactor shutdown systems with high reliability ensures that at least one will operate following any single process failure. The ECI system as the heat sink is designed to prevent failure of the fuel from overheating. ECI or moderator systems are capable of maintaining the integrity of the fuel channels for all design basis accidents.

Bruce A and Bruce B design and operation aligns with the 37 Safety Principles related to Level 3 defence-in-depth. Improvements identified in 49 GIOs that are included in the IIP will further enhance Level 3 DID.

18.4.4. Level 4

The aim of the fourth level of defence-in-depth is to ensure that radioactive releases caused by severe accidents are kept as low as practicable.

Implementation of SAMG provides for equipment and procedures to manage response to low probability severe accidents and mitigate their consequences as far as practicable.

Adequate protection for the confinement function is provided by way of a robust containment design and implementation of design improvements based on the Fukushima follow-up actions. The confinement function is further enhanced by severe accident management procedures and improvements to SAM guidance based on Fukushima follow-up actions.

Bruce A and Bruce B design and operation aligns with the 33 Safety Principles related to Level 4 defence-in-depth. Improvements identified in 45 GIOs that are included in the IIP will further enhance Level 4 DID.

18.4.5. Level 5

The aim of the fifth level of defence is to mitigate the radiological consequences of potential releases of radioactive materials that may result from accident conditions.

Bruce A and Bruce B design and operation provides adequately equipped emergency support facilities, trained staff and plans for onsite and offsite emergency response.

Bruce Power has in place all the requisite governance, implementing procedures, facilities, equipment and specifically staff trained to support Province of Ontario in managing and mitigating off-site radiological consequences in the event of a nuclear accident.

Bruce Power is implementing further improvements to emergency planning governance and facilities based on the Fukushima follow-up actions.

Bruce A and Bruce B design and operation aligns with the 7 Safety Principles related to Level 5 defence-in-depth. Improvements identified in 2 GIOs that are included in the IIP will further enhance Level 5 DID.

18.4.6. Summary

In summary, there are effective and adequate provisions in place for all levels of defence-in-depth in Bruce A and Bruce B. Moreover, improvement opportunities that have been included in the IIP based on the results of the PSR and GA described in this report will further improve the defence-in-depth provisions of Bruce A and Bruce B.

19. Assessment of Overall Safety

This Global Assessment shows that the overall risk associated with operation of Bruce A and Bruce B over the designated PSR period is acceptably low. This conclusion is based on the following key elements of the Global Assessment.

1. **Significant physical improvements have been implemented by Bruce Power since the Bruce A Units were returned to operation (B1&2 in 2012, B3 in 2004, B4 in 2003) and over the operating life of Bruce B units to date**

Over the lifetime of the Bruce A and Bruce B units major projects have been completed to improve the physical plant to meet PROL conditions, align plant design with modern codes and standards, enhance defence-in-depth provisions and to maintain and improve safety margins. Some of the notable major projects that have been implemented are:

- Special safety system design changes and equipment upgrades as part of their return to service (Units 1 and 2)
- Replacement of Pressure Tubes and Calandria Tubes (Units 1 and 2)
- Replacement of Steam Generators with improved materials and designs (Units 1 and 2)
- Replacement of portions of feeders with improved materials and designs (Units 1 and 2)
- Fuel Handling upgrades (Units 1 and 2)
- Heat Transport Pump Transformer protection circuit upgrades (Unit 3)
- Station electrical power transformer replacements (Units 0A, 3 and 4)
- Pressure tube elongation life extension (West-shift-plus) (Unit 3)
- Nuclear instrument upgrades (Units 3 and 4)
- Liquid Zone Control pump replacement (Unit 4)

- Low pressure turbine replacement (Units 1 and 4)
- Generator replacement (Unit 4)
- Qualified Power Supply (Bruce A)
- Implementation of seismic qualification and installation of seismic modifications (Bruce A)
- Installation of Maintenance Cooling vent lines (Units 5 and 7)
- Valve and pump replacements in the Steam and Feedwater Heating systems (Unit 6)
- Heat Transport System solid mode pressure control system upgrade (Bruce B)
- Battery Bank Replacements (Bruce B)
- Standby Generator 7 and 8 control system upgrades (Bruce B)
- PHT Pump Motor refurbishment (Bruce B)
- Controller replacements throughout the station (Bruce B)
- Motor Control Centre refurbishments (Bruce B)
- Digital Control Computer air conditioning units (Bruce B)
- Implementation of Environmental Qualification Project and design upgrades (Bruce A and B)
- Installation of a secondary control area (Bruce A)
- Design and operational enhancements as part of Fukushima Actions (Bruce A and B):
  - Installation of PARs to improve hydrogen control for beyond design basis accident conditions
  - Power supply and cooling capability for beyond design basis accident conditions
  - SAMG enhancements to address multi-unit severe accidents
  - New Emergency Response facilities
  - Installation of automated real-time station boundary monitoring equipment, including 44 gamma detectors (16 on-site detectors with the remaining 28 within 10 km area around the site).
- Implementation of new fire protection requirements and installation of fire protection upgrades
- Bruce A and Bruce B Core conversion to fueling with flow to improve LLOCA safety margins (Bruce A and B)
- Implementation of 37M Fuel Bundles to improve CHF margin (Bruce A and B)

As demonstrated in Section 18, Bruce A and B design and operation is aligned with all the safety principles associated with all 5 levels of DID. Looking forward, the IIP described in

Sections 15 and 16 will result in further improvements in all levels of DID, reducing the overall risk associated with the operation of the plant.

2. The IIP presented in Appendix A, plus major projects and initiatives driven by Asset Life Management and REGDOC-2.6.3 implementation will mitigate SSC ageing and improve safety margins

Improvement in safety margins associated with IIP implementation cannot be accounted for quantitatively without a supporting safety analysis. The IIP includes the Safety Report improvement initiative under GIO-009, which addresses CNSC AI 090739. It is expected that on-going analyses under this initiative will credit safety improvements as appropriate and demonstrate adequate safety margins over the PSR period. Qualitatively, defence-in-depth reviews have shown that the current IIP will enhance all levels of DID, which will allow management of safety margins associated with aging and improve safety margins.

Major component replacements for life extension have been completed in Units 1 and 2 to manage and recover safety margins associated with the ageing of SSCs and to improve plant safety and reliability beyond the current PSR period. A decision on the timing of MCR has been made and Bruce Power's plans to extend the life of Units 3 to 8 have been communicated to the CNSC [9] [10]. Other initiatives that will assure safe and reliable plant operation beyond the current PSR will be integrated in the planned outages or MCR outages and included in future updates of the IIP, as appropriate.

It should also be emphasized that the scope of improvement initiatives as part of on-going operations is much wider than those that have to be included in the IIP as part of the PSR process. Asset life management initiatives listed in Table 31 in Section 9.3 of this report demonstrate the subset of the initiatives that will be performed as part of MCR outages.

3. Continued compliance with regulatory dose limits as well as Bruce Power's safety goals

Deterministic and probabilistic safety analysis results demonstrate that Bruce A and B operation over the designated licence period meets the regulatory requirements, as well as Bruce Power's safety goals, with significant margins and hence risks associated with radioactive hazards will continue to remain acceptably low. In addition, processes are in place to update the safety analysis to quantify the impact of design and operational improvements to the plant, as well as the impact of SSC ageing to demonstrate continuous compliance with the deterministic and probabilistic risk acceptance criteria associated radioactive hazards.

Deterministic Safety Analysis

Results of deterministic safety analysis are documented in the plant Safety Reports. The DSA demonstrates that the radiological consequences of the accidents analyzed involving a single process failure and a single process failure in conjunction with failure of one of the special safety systems do not exceed the public dose limits specified in the Siting Guide and in most cases with significant margins. These limits, extracted from the LCH [6], are reproduced below.

	Individual Dose Limit		Population Dose Limit	
	Thyroid Dose (mSv)	Whole Body Dose (mSv)	Thyroid Dose (Person mSv)	Whole Body Dose (Person mSv)
Single Failure	30	5	10^5	10^5
Dual Failure	2500	250	10^7	10^7

Bruce Power also conducted safety analysis to address the impact of aging on safety margins for the 2015-2020 licence period. The analysis results demonstrate that Bruce A and B have adequate safety margins and continues to meet the public dose acceptance criteria and in most cases with significant margins. MCR outages planned during the 10-year PSR period will recover the safety margins associated with the ageing of Units 3 to 8 for extended life.

Probabilistic Safety Assessment

Bruce A and Bruce B PSA demonstrates that the current plant meets the established safety goals. Bruce Power safety goals are:

- Quantitative Safety Goal for Severe Core Damage Frequency (SCDF): Sum of frequencies of all event sequences that can lead to significant core degradation should not exceed 10^{-4} occurrences per reactor-year;
- Quantitative Safety Goal for Small Release Frequency (SRF): Sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{15} Becquerels of Iodine-131 should not exceed 10^{-4} occurrences per reactor-year;
- Quantitative Safety Goal for Large Release Frequency (LRF): Sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{14} Becquerels of Cesium-137 should not exceed 10^{-5} occurrences per reactor-year.

The results of the latest Bruce A and Bruce B PSAs individually meet all of Bruce Power's probabilistic safety goals. These results also demonstrate significant risk reduction due to installation of Emergency Mitigating Equipment as a result of the Fukushima-related improvement initiatives.

4. Impact of those findings that were not included for consideration in the IIP

As summarized in Section 8.2.1, there are 74 sets of micro-gaps that were classified as 'impracticable'.

- Fifty-four sets are those associated with physical design originating from a clause-by-clause review of REGDOC-2.5.2 Design of Reactor Facilities: Nuclear Power Plants.
- Twenty sets are those associated with physical design originating from a clause-by-clause or high level review of CSA standards.

These modern regulatory documents, codes and standards deal with a wide variety of topics related to requirements suited to new NPPs. Specifically for REGDOC-2.5.2, the general nuclear safety objective, which is supported by three complementary safety objectives on radiation protection, technical safety and environmental safety, has quantitatively more conservative requirements and associated limits as compared to those prescribed in the PROL for Bruce A and Bruce B. The technical safety objectives are to provide all reasonably practicable measures to prevent accidents in the NPP, and to mitigate the consequences of accidents if they do occur. This takes into account all possible accidents considered in the design, including those of very low probability. Although these objectives, their associated technical requirements, dose acceptance criteria and safety goals were not in effect at the time of the original design of Bruce A and Bruce B, significant design improvements have been made over the years to align with modern codes and standards where practicable.

Resolution of these micro-gaps requires a fundamentally different approach that would affect the plant design as a whole and SSC design specifically. In addition, when taken as a whole, the number of impracticable micro-gaps that require fundamental design changes to specific SSCs and the plant as a whole, and their integrated impact with potentially conflicting physical and layout constraints, cannot be accommodated within the current configuration of SSCs and plant layout. Due to the existing coupling of SSCs and their functional capabilities in the current design of the plant, changes to SSC(s) would also impact other physically connected or functionally related SSCs. This integrated impact will further increase the level of complexity and impracticability.

Bruce A and B meet the current deterministic safety analysis dose acceptance limits, as well as the probabilistic safety assessment safety goals, and in most cases with significant safety margins. Therefore, impacts of these ‘impracticable’ micro-gaps are not considered to be safety significant for the current PSR interval.

These micro-gaps will be revisited in the next update of the PSR.

5. Summary and Conclusions

As demonstrated by safety analysis, design and operation of Bruce A and Bruce B meet the current deterministic safety analysis dose acceptance limits, as well as the probabilistic safety assessment safety goals with significant safety margins. Implementation of IIP initiatives will maintain or further improve the associated safety margins and assure continued overall safety for the PSR period. Therefore, the overall conclusion is that ongoing safe operation of Bruce A and Bruce B for the PSR period is assured through the plant design, governance and improvements listed in the IIP.

20. Justification for Proposed Continued Operation

Continued operation of Bruce A and Bruce B over the designated PSR period is acceptable based on the results of the Global Assessment and the improvements that will be implemented through the IIP. This is supported by the following:

- A comprehensive PSR of Bruce A and Bruce B has been completed. This review covered the current organization, governance and processes associated with all aspects of plant design, operation and condition of the physical plant against the current licensing basis, as well as modern codes and standards. No immediate safety concerns have been identified.
- The extent to which Bruce A and Bruce B currently meet new requirements that may become part of the licensing basis in the future has been assessed and practicable improvement opportunities have been included in the IIP. These improvement opportunities will further enhance safe and reliable operation and align Bruce A and Bruce B design and operation with modern regulatory documents, codes and standards applicable to new NPPs.
- Adequacy of the design and operation in terms of DID has been demonstrated, including compliance with the fundamental safety principles associated with all 5 levels of DID.
- Design and operation of the plants meet the current deterministic safety analysis dose acceptance limits of the PROL, as well as Bruce Power's probabilistic safety analysis safety goals, in most cases with significant safety margins. Processes are in place to update the safety analysis as required to take ageing into account. Therefore, the overall risk associated with operation of Bruce A and Bruce B over the designated PSR period is acceptably low.
- A framework, as shown in Figure 2, has been put in place that integrates improvements planned or in-progress based on asset life management and safety basis review inputs and those proposed in the IIP to mitigate SSC aging and enhance current safety margins for continued safe and reliable long-term operation.
- The improvements described in the IIP will resolve the associated micro-gaps to enhance safe and reliable operation of Bruce A and Bruce B over the designated PSR period and beyond.
- Bruce Power's current organizational structure and management system provides the requisite processes, tools, resources and oversight that will ensure effective execution of the IIP.

Part VI: Supporting Documentation

Section	Title
21	References

Appendix	Title

No supporting Appendices required.

21. References

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Appendix A – Integrated Implementation Plan

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
Physical design	2014	GIO-001	Improve documented design basis	Bruce B	CA-0006	SIP-13B: BB Legacy Registration	0.0161	31-Dec-17	NK29-CORR-00531-13701 NK29-CORR-00531-12884 NK29-CORR-00531-11687 AI 091413
				Bruce A & Bruce B	CA-0191	Update governing procedures and implementing documents on seismic qualification	0.0161	18-Dec-20	
Physical design	2014	GIO-002	Implement design changes to improve severe accident response	Bruce A	CA-0009	SIP-1A: Fukushima Response - Bruce A External Water Makeup to Heat Transport System and Moderator System	0.0008	20-Dec-19	NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-11298 / NK29-CORR-00531-11708 / NK37-CORR-00531-02229 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 AI 2014-07-3688
				Bruce B	CA-0010	SIP-1B: Fukushima Response - Bruce B External Water Makeup to Heat Transport System and Moderator System	0.0008	18-Dec-20	NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-11298 /

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
									NK29-CORR-00531-11708 / NK37-CORR-00531-02229 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 AI 2014-07-3688
				Bruce A	CA-0011	SIP-2A: Fukushima Response - Bruce A Containment Venting Connection Point and Passive CFVS Installation	0.0008	30-Mar-18	B-REP-34310-00002 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-12417 / NK29-CORR-00531-12829 / NK37-CORR-00531-02474 AI 2015-07-3683 NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254
				Bruce B	CA-0012	SIP-2B: Fukushima Response Bruce B - Containment Venting Connection Point and Passive CFVS Installation	0.0008	30-Mar-18	B-REP-34310-00002 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-12417 / NK29-CORR-00531-12829 / NK37-CORR-00531-02474 NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
									AI 2015-07-3683 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 B-REP-34310-00002
				Bruce A	CA-0013	SIP-4: Fukushima Response (SAMG Improvement) - Bruce A Wide range ECI Sump Level Indication	0.0008	21-Dec-18	NK21-CORR-00531-12282 NK21-CORR-00531-12123
Physical design	2014	GIO-003	Assess pipe whip and jet impingement	Bruce B	CA-0192	SF1-3: Perform an assessment of pipe whip and jet impingement	0.0027	18-Dec-20	NK21-CORR-00531-12191 NK21-CORR-00531-11567 / NK29-CORR-00531-11950/ NK37-CORR-00531-02288 NK21-CORR-00531-08706
Physical design	2014	GIO-005	Assess cyclic loads of pressure retaining components designed per ASME III or VIII	Bruce B	CA-0028	SF1-8: Evaluate impact of fatigue due to cyclic operation transient loads on Class 4 Containment Penetrations	0.0027	18-Dec-20	NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457
				Bruce B	CA-0029	SF1-9: Evaluate impact of fatigue for Class 2, 3 and 4 bellows expansion joints	0.0027	18-Dec-20	NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457
				Bruce B	CA-0030	SF1-12: Evaluate Class 6 piping components for cyclic and dynamic reactions	0.0027	18-Dec-20	NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457
Safety analysis	2014	GIO-009	Update safety analysis to align with REGDOC-2.4.1	Bruce A & Bruce B	CA-0043	SIP-3:REGDOC-2.4.1 Implementation	0.0646	22-Dec-17	NK21-CORR-00531-12334 / NK29-CORR-00531-12767 NK21-CORR-00531-10774 NK29-CORR-00531-11155 NK21-CORR-00531-11214

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
									NK29-CORR-00531-11621 AI 090739
				Bruce A & Bruce B	CA-0174	Safety Report & Probabilistic Safety Assessment	0.0646	23-Dec-22	
Operating performance	2014	GIO-011	Implement enhancements to SAMG	Bruce A & Bruce B	CA-0047	SIP-11: Fukushima Response - Severe Accident Management Enhancements	0.0039	18-Dec-20	NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-11801 / NK29-CORR-00531-12195 / NK37-CORR-00531-02338 NK21-CORR-00531-12554 / NK29-CORR-00531-12979 / NK37-CORR-00531-02511 2014-07-3688 NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254
Physical design	2014	GIO-019	Assess and improve seismic qualification	Unit 1 & 2	CA-0061	SIP-16: BA U1/U2 Post RTS - Seismic Margin Upgrade (IIP-6)	0.1475	30-Mar-18	NK21-CORR-00531-13426 NK21-CORR-00531-12257 NK21-CORR-00531-12647 NK21-CORR-00531-11170 AI 1407-4602
Management system	2014	GIO-024	Enhanced Periodic Safety Review to Support Asset Management	Bruce A & Bruce B	CA-0066	SIP-22: Enhanced Periodic Safety Review to Support Asset Management	0.0006	22-Dec-17	NK21-CORR-00531-12269 NK21-CORR-00531-10576 NK29-CORR-00531-10975

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
Fitness for service	2014	GIO-025	Perform R&D in support of fuel channel life cycle management initiatives	Bruce A & Bruce B	CA-0067	Fuel Channel Life Management	0.4916	20-Dec-19	NK21-CORR-00531-13680 / NK29-CORR-00531-14326 CNSC e-Docs #5243387-v3 NK21-CORR-00531-10978 / NK29-CORR-00531-11366 NK21-CORR-00531-11472 NK21-CORR-00531-12248 / NK29-CORR-00531-12672 NK21-CORR-00531-12618 / NK29-CORR-00531-13046 NK21-CORR-00531-12662 / NK29-CORR-00531-13098 NK21-CORR-00531-12921 / NK29-CORR-00531-13384 NK21-CORR-00531-13019 / NK29-CORR-00531-13486 NK21-CORR-00531-13380 / NK29-CORR-00531-13927 NK21-CORR-00531-13414 / NK29-CORR-00531-13961 NK29-CORR-00531-11868 AI 1407-4775
Physical design	2014	GIO-026	BA & BB New Neutronic Trips	Bruce A & Bruce B	CA-0069	SIP-25: BA & BB New Neutronic Trips Feasibility Project	0.0008	20-Dec-19	NK21-CORR-00531-12850 / NK29-CORR-00531-13310 NK21-CORR-00531-12491 / NK29-CORR-00531-12909 AI 1207-3320 NK21-CORR-00531-11357 NK29-CORR-00531-11762 AI 1207-3320

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
Fitness for service	2014	GIO-028	Upgrade Emergency and Standby Power Supplies	Bruce A	CA-0071	SIP-30: BA U1/U2 Post RTS - Standby Generator Controls Replacement	0.4916	17-Dec-21	NK21-CORR-00531-13161 / NK29-CORR-00531-13649 NK21-CORR-00531-12449 / NK29-CORR-00531-12861 NK21-CORR-00531-11366 AI 1207-3283
				Bruce B	CA-0073	SIP-35: Emergency Power Generators 1 and 2 Upgrades	0.4916	22-Dec-17	NK29-CORR-00531-13479 NK29-CORR-00531-12003 NK29-CORR-00531-12077 NK29-CORR-00531-09598 AI 111402
Fitness for service	2015	GIO-034	Safety System Reliability	Bruce A	CA-0078	Improvement of unavailability targets for some safety related systems	0.2263	21-Dec-18	
Physical design	2015	GIO-036	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment	Bruce A & Bruce B	CA-0080	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment	0.0027	18-Dec-20	
Physical design	2015	GIO-037	Document design basis for zoning and shielding	Bruce A & Bruce B	CA-0081	Establish technical basis for radiation zone designation	0.0027	20-Dec-19	
				Bruce A & Bruce B	CA-0082	Shielding design criteria and the methodology for specification of shielding parameters and material selection	0.0027	20-Dec-19	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
Fitness for service	2015	GIO-039	Equipment Reliability and Maintenance	Bruce A & Bruce B	CA-0084	In-Service Inspection Program for Bruce NGS A and B Safety Related Structures	0.3460	20-Dec-19	
Human performance management	2015	GIO-043	Validation of Human Credited Actions	Bruce A & Bruce B	CA-0089	Validation of human actions credited under accident conditions in the safety report	0.0322	20-Dec-19	
				Bruce A & Bruce B	CA-0177	Definition of staff availability requirements for supporting heat sink availability	0.0037	21-Dec-18	
Emergency management and fire protection	2015	GIO-044	Emergency preparedness	Bruce A & Bruce B	CA-0090	Emergency response documentation	0.0024	21-Dec-18	
				Bruce A & Bruce B	CA-0199	Complete the On-Site/Off-Site Emergency Response Communications Project	0.0022	20-Dec-19	
				Bruce A & Bruce B	CA-0200	Addressing outstanding follow-up actions from Audits on Emergency Preparedness	0.0048	21-Dec-18	
				Bruce A & Bruce B	CA-0201	Address the following issues identified as part of the OSART review with respect to ERP	0.0182	21-Dec-18	
Fitness for service	2016	GIO-056	Fuel Channel Replacement	Unit 6	CA-0120	Fuel Channel Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0209	Fuel Channel Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0226	Fuel Channel Replacement - Unit 4	0.4916	31-Dec-27	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 5	CA-0243	Fuel Channel Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0260	Fuel Channel Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0277	Fuel Channel Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-057	Steam Generator Replacement	Unit 6	CA-0121	Steam Generator Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0210	Steam Generator Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0227	Steam Generator Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0244	Steam Generator Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0261	Steam Generator Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0278	Steam Generator Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-058	Feeder Replacement	Unit 6	CA-0122	Feeder Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0211	Feeder Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0228	Feeder Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0245	Feeder Replacement - Unit 5	0.4916	30-Jun-29	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 7	CA-0262	Feeder Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0279	Feeder Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-059	Calandria and Shield Tank Assembly Major Inspection	Unit 6	CA-0123	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 6	0.3460	31-Dec-23	
				Unit 3	CA-0212	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 3	0.3460	30-Jun-26	
				Unit 4	CA-0229	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 4	0.3460	31-Dec-27	
				Unit 5	CA-0246	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 5	0.3460	30-Jun-29	
				Unit 7	CA-0263	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 7	0.3460	30-Jun-31	
				Unit 8	CA-0280	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 8	0.3460	30-Jun-33	
Fitness for service	2016	GIO-060	Preheater Inspections	Unit 6	CA-0124	Preheater Inspections - Unit 6	0.3460	31-Dec-23	
				Unit 3	CA-0213	Preheater Inspections - Unit 3	0.3460	30-Jun-26	
				Unit 4	CA-0230	Preheater Inspections - Unit 4	0.3460	31-Dec-27	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 5	CA-0247	Preheater Inspections - Unit 5	0.3460	30-Jun-29	
				Unit 7	CA-0264	Preheater Inspections - Unit 7	0.3460	30-Jun-31	
				Unit 8	CA-0281	Preheater Inspections - Unit 8	0.3460	30-Jun-33	
Fitness for service	2016	GIO-062	PHT Pump Seal Bellows Replacement	Unit 6	CA-0126	PHT Pump Seal Bellows Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0215	PHT Pump Seal Bellows Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0232	PHT Pump Seal Bellows Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0249	PHT Pump Seal Bellows Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0266	PHT Pump Seal Bellows Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0283	PHT Pump Seal Bellows Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-064	Control Distribution Frame (CDF) Terminal Replacement	Unit 3	CA-0334	Control Distribution Frame (CDF) Terminal Replacement - Unit 3	0.0702	30-Jun-26	
				Unit 4	CA-0335	Control Distribution Frame (CDF) Terminal Replacement - Unit 4	0.0702	31-Dec-27	
Fitness for service	2016	GIO-065	PHT Seismic Restraints (Snubbers)-	Unit 6	CA-0129	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-	0.3460	31-Dec-23	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
			Periodic Inspection Program (PIP)-Inspection			Inspection - Unit 6			
				Unit 3	CA-0336	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 3	0.3460	30-Jun-26	
				Unit 4	CA-0337	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 4	0.3460	31-Dec-27	
				Unit 5	CA-0338	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 5	0.3460	30-Jun-29	
				Unit 7	CA-0339	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 7	0.3460	30-Jun-31	
				Unit 8	CA-0340	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 8	0.3460	30-Jun-33	
Fitness for service	2016	GIO-066	Pressurizer and Supports- Internal Inspection	Unit 6	CA-0130	Pressurizer and Supports-Internal Inspection - Unit 6	0.3460	31-Dec-23	
				Unit 5	CA-0341	Pressurizer and Supports-Internal Inspection - Unit 5	0.3460	30-Jun-29	
				Unit 7	CA-0342	Pressurizer and Supports-Internal Inspection - Unit 7	0.3460	30-Jun-31	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 8	CA-0343	Pressurizer and Supports-Internal Inspection - Unit 8	0.3460	30-Jun-33	
				Unit 3	CA-0344	Pressurizer and Supports-Internal Inspection - Unit 3	0.3460	30-Jun-26	
				Unit 4	CA-0345	Pressurizer and Supports-Internal Inspection - Unit 4	0.3460	31-Dec-27	
Fitness for service	2016	GIO-070	Air Operated Valves-Replacement	Unit 6	CA-0138	Air Operated Valve- Nuclear Valve Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 6	CA-0139	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0217	Air Operated Valve- Nuclear Valve Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0234	Air Operated Valve- Nuclear Valve Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0251	Air Operated Valve- Nuclear Valve Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0268	Air Operated Valve- Nuclear Valve Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0285	Air Operated Valve- Nuclear Valve Replacement - Unit 8	0.4916	30-Jun-33	
				Unit 3	CA-0329	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0330	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 4	0.4916	31-Dec-27	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 5	CA-0331	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0332	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0333	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-071	Large Motors-Refurbishment/Replacement	Unit 6	CA-0145	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0346	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0347	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0348	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 5	0.4916	30-Jun-29	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 7	CA-0352	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0353	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-076	DCC Cables and WIBAs – Replacement	Unit 6	CA-0153	DCC Cables and WIBAs - Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0221	DCC Cables and WIBAs - Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0238	DCC Cables and WIBAs - Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0255	DCC Cables and WIBAs - Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0272	DCC Cables and WIBAs - Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0289	DCC Cables and WIBAs - Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-077	Moderator Heat Exchangers-Replacement	Unit 6	CA-0154	Moderator Heat Exchangers-Replacement - Unit 6	0.4916	23-Dec-33	
				Unit 3	CA-0222	Moderator Heat Exchangers-Replacement - Unit 3	0.4916	30-Jun-26	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 4	CA-0239	Moderator Heat Exchangers-Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0256	Moderator Heat Exchangers-Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0273	Moderator Heat Exchangers-Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0290	Moderator Heat Exchangers-Replacement - Unit 8	0.4916	30-Jun-33	
Fitness for service	2016	GIO-078	Maintenance Cooling Heat Exchanger-Replacement	Unit 6	CA-0155	Maintenance Cooling Heat Exchanger- Replacement - Unit 6	0.4916	31-Dec-23	
				Unit 3	CA-0223	Maintenance Cooling Heat Exchanger- Replacement - Unit 3	0.4916	30-Jun-26	
				Unit 4	CA-0240	Maintenance Cooling Heat Exchanger- Replacement - Unit 4	0.4916	31-Dec-27	
				Unit 5	CA-0257	Maintenance Cooling Heat Exchanger- Replacement - Unit 5	0.4916	30-Jun-29	
				Unit 7	CA-0274	Maintenance Cooling Heat Exchanger- Replacement - Unit 7	0.4916	30-Jun-31	
				Unit 8	CA-0291	Maintenance Cooling Heat Exchanger- Replacement - Unit 8	0.4916	30-Jun-33	
Physical design	2015	GIO-081	Human Factors in Design of Nuclear Power Plants	Bruce A & Bruce B	CA-0179	HF design review of control room, workstations, computer interfaces, alarms	0.0027	21-Dec-18	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
						systems, soft control systems, communication systems and field components relevant to safety			
Environmental protection	2015	GIO-082	Performance testing of nuclear air-cleaning systems	Bruce A & Bruce B	CA-0168	Air pressure measurements in support of emission estimates	0.2263	21-Dec-18	
				Bruce A & Bruce B	CA-0169	QA/QC guidance for performance testing of nuclear air-cleaning systems	0.2263	21-Dec-18	
				Bruce A & Bruce B	CA-0170	Effectiveness reviews of the air-cleaning system performance testing program	0.2263	21-Dec-18	
				Bruce A & Bruce B	CA-0171	Requirements for the qualifications of personnel who conduct air filter performance testing	0.2263	21-Dec-18	
				Bruce A & Bruce B	CA-0172	Performance testing of nuclear air-cleaning systems-Program documentation	0.2263	21-Dec-18	
				Bruce A & Bruce B	CA-0173	Pre-service and in-service testing of adsorbent media (activated carbon)	0.2263	21-Dec-18	
Safety analysis	2016	GIO-083	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2	Bruce A & Bruce B	CA-0190	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2	0.0017	18-Dec-26	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
Fitness for service	2016	GIO-086	PHT Valves-Refurbishment of 33120-MV23	Unit 6	CA-0207	PHT Valves-Refurbishment of 33120-MV23 - Unit 6	0.3460	31-Dec-23	
				Unit 3	CA-0224	PHT Valves-Refurbishment of 33120-MV23 - Unit 3	0.3460	30-Jun-26	
				Unit 4	CA-0241	PHT Valves-Refurbishment of 33120-MV23 - Unit 4	0.3460	31-Dec-27	
				Unit 5	CA-0258	PHT Valves-Refurbishment of 33120-MV23 - Unit 5	0.3460	30-Jun-29	
				Unit 7	CA-0275	PHT Valves-Refurbishment of 33120-MV23 - Unit 7	0.3460	30-Jun-31	
				Unit 8	CA-0292	PHT Valves-Refurbishment of 33120-MV23 - Unit 8	0.3460	30-Jun-33	
Management system	2016	GIO-088	Improve Licencing Processes	Bruce A & Bruce B	CA-0294	Licence Concessions Database	0.2267	20-Dec-19	
Safety analysis	2016	GIO-089	Whole-Site Probabilistic Risk Assessment	Bruce A & Bruce B	CA-0376	Development and Implementation of Whole-Site Probabilistic Risk Assessment	0.0986	20-Dec-19	NK21-CORR-00531-11715 / NK29-CORR-00531-12105 NK21-CORR-00531-12837 / NK29-CORR-00531-13287 NK21-CORR-00531-12973 / NK29-CORR-00531-13444 NK21-CORR-00531-13030 / NK29-CORR-00531-13499
Physical design	2016	GIO-090	SDS2 Enhancements	Unit 3	CA-0297	Implement SDS2 Neutron Overpower Protection Enhancements - Unit 3	0.0008	30-Jun-26	
				Unit 4	CA-0378	Implement SDS2 Neutron Overpower Protection Enhancements - Unit 4	0.0008	31-Dec-27	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
Physical design	2016	GIO-091	Bruce A Fire Protection Upgrades to Align with CSA-N293-07	Bruce A	CA-0299	BA ASB Fire Protection Upgrades	0.0008	22-Dec-17	NK21-CORR-00531-13173 / NK29-CORR-00531-13659
				Units 1 and 2	CA-0300	Unit 1 and 2 Fire Upgrades (Restart - Project #38730)	0.0008	30-Jun-20	NK21-CORR-00531-13031
				Bruce A	CA-0301	BA Standby Generator Building Fire Protection Upgrade	0.0008	17-Dec-21	NK21-CORR-00531-13173 / NK29-CORR-00531-13659
				Bruce A	CA-0302	Bruce A Fire Barriers Upgrades (Cable Wraps)	0.0008	18-Dec-20	NK21-CORR-00531-13173 / NK29-CORR-00531-13659
				Bruce A	CA-0303	Bruce A Very Early Smoke Detection Apparatus (VESDA) Upgrade	0.0008	20-Dec-19	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955 NK21-CORR-00531-11324 / NK29-CORR-00531-11729
				Units 1 and 2	CA-0304	Unit 1 and 2 Fire Upgrades (SCA VESDA & Turbine Sprinkler System alarm detection and notification interface)	0.0008	18-Dec-20	NK21-CORR-00531-13031
Physical design	2016	GIO-092	Bruce B Fire Protection Upgrades to Align with CSA-N293-07	Unit 0B	CA-0306	BB U0 Fuel Storage Area Sprinkler Upgrades	0.0008	20-Dec-19	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
				Bruce B	CA-0307	Bruce B Fireworks Terminal Replacement	0.0008	20-Dec-19	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Bruce B	CA-0308	Bruce B Firewater Pipe Replacement	0.0008	18-Dec-20	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
				Bruce B	CA-0309	Bruce B Fire Detection Upgrade	0.0008	17-Dec-21	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
				Bruce B	CA-0310	Bruce B Very Early Smoke Detection Apparatus (VESDA) Upgrade	0.0008	18-Dec-20	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955 NK21-CORR-00531-11324/ NK29-CORR-00531- 11729
				Bruce B	CA-0311	Bruce B Fire Barriers (Cable Wrap) upgrades	0.0008	18-Dec-20	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
				Bruce B	CA-0312	Bruce B Standby Generator Building Fire Protection Upgrade	0.0008	17-Dec-21	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
				Bruce B	CA-0313	BB EPG / EWPS Building Fire Protection Upgrade	0.0008	17-Dec-21	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
				Unit 8	CA-0315	Unit 8 Fire Upgrades	0.0008	18-Dec-20	NK21-CORR-00531-13173 / NK29-CORR-00531-13659

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Bruce A & Bruce B	CA-0316	Air Foam System Replacement	0.0008	18-Dec-20	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955
Radiation protection	2015	GIO-093	RP equipment and instrumentation maintenance and life cycle management	Bruce A & B	CA-0317	RP Instrumentation life cycle management	0.0201	20-Dec-19	
				Bruce A & B	CA-0318	RP Instrumentation maintenance	0.0306	20-Dec-19	
				Bruce A & B	CA-0320	Technical Basis for RP instrumentation setpoints, locations and function checks	0.0201	21-Dec-18	
Radiation protection	2015	GIO-094	Effective use of the action tracking system in Radiation Protection	Bruce A & Bruce B	CA-0319	Improve effective use of the action tracking system in Radiation Protection	0.0201	21-Dec-18	
Fitness for service		GIO-095	45VDC Power Supplies-Replacement	Unit 0A	CA-0321	45VDC Power Supplies-Replacement - Unit 0A	0.3460	31-Dec-27	
				Unit 0B	CA-0322	45VDC Power Supplies-Replacement - Unit 0B	0.3460	30-Jun-33	
				Unit 3	CA-0323	45VDC Power Supplies-Replacement - Unit 3	0.3460	30-Jun-26	
				Unit 4	CA-0324	45VDC Power Supplies-Replacement - Unit 4	0.3460	31-Dec-27	
				Unit 5	CA-0325	45VDC Power Supplies-Replacement - Unit 5	0.3460	30-Jun-29	
				Unit 6	CA-0326	45VDC Power Supplies-Replacement - Unit 6	0.3460	31-Dec-23	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 7	CA-0327	45VDC Power Supplies-Replacement - Unit 7	0.3460	30-Jun-31	
				Unit 8	CA-0328	45VDC Power Supplies-Replacement - Unit 8	0.3460	30-Jun-33	
Physical design		GIO-097	Bruce A Legacy Registration-Implementation Projects	Bruce A	CA-0298	Documentation - Legacy Registration Project DCN/DCPs- Bruce A	0.0004	31-May-17	NK21-CORR-00531-11941 NK21-CORR-00531-09328 NK21-CORR-00531-08728 NK21-CORR-00531-08217 NK21-CORR-00531-05602
				Bruce A	CA-0349	Implementation - Legacy Registration Project DCN/DCPs- Bruce A	0.0008	21-Dec-18	
Physical design		GIO-098	Bruce B Legacy Registration-Implementation Projects	Bruce B	CA-0351	Implementation - Legacy Registration Project DCN/DCPs- Bruce B	0.0008	21-Dec-18	
Physical design	2016	GIO-099	Install Correctly Sized Maintenance Cooling Relief Valves	Bruce B	CA-0314	BB Maintenance Cooling Interspace Protection	0.0350	21-Dec-18	NK29-CORR-00531-14091 NK29-CORR-00531-13950
Physical design	2016	GIO-100	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications	Unit 3	CA-0354	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 3	0.4916	21-Dec-18	
				Unit 4	CA-0355	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 4	0.4916	21-Dec-18	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 5	CA-0356	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 5	0.4916	21-Dec-18	
				Unit 6	CA-0357	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 6	0.4916	21-Dec-18	
				Unit 7	CA-0358	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 7	0.4916	21-Dec-18	
				Unit 8	CA-0359	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 8	0.4916	21-Dec-18	
Physical design	2016	GIO-101	M/34720 Relief Valves For Overpressure Protection	Unit 1	CA-0360	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 1	0.4916	21-Dec-18	
				Unit 2	CA-0361	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 2	0.4916	21-Dec-18	
				Unit 3	CA-0362	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 3	0.4916	21-Dec-18	
				Unit 4	CA-0363	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 4	0.4916	21-Dec-18	
				Unit 5	CA-0364	M/34720 Replacement of Relief Valves For	0.4916	21-Dec-18	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
						Overpressure Protection - Unit 5			
				Unit 6	CA-0365	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 6	0.4916	21-Dec-18	
				Unit 7	CA-0366	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 7	0.4916	21-Dec-18	
				Unit 8	CA-0367	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 8	0.4916	21-Dec-18	
Physical design	2016	GIO-102	I/63472 Remote Relief Valve Position Indication	Unit 1	CA-0368	I/63472 Remote Relief Valve Position Indication - Unit 1	0.4916	21-Dec-18	
				Unit 2	CA-0369	I/63472 Remote Relief Valve Position Indication - Unit 2	0.4916	21-Dec-18	
				Unit 3	CA-0370	I/63472 Remote Relief Valve Position Indication - Unit 3	0.4916	21-Dec-18	
				Unit 4	CA-0371	I/63472 Remote Relief Valve Position Indication - Unit 4	0.4916	21-Dec-18	
				Unit 5	CA-0372	I/63472 Remote Relief Valve Position Indication - Unit 5	0.4916	21-Dec-18	
				Unit 6	CA-0373	I/63472 Remote Relief Valve Position Indication - Unit 6	0.4916	21-Dec-18	

CNSC S&C Area	IIP	GIO No.	GIO TITLE	Applicable Unit(s)	CARD #	CARD TITLE	CARD Score	TCD	References
				Unit 7	CA-0374	I/63472 Remote Relief Valve Position Indication - Unit 7	0.4916	21-Dec-18	
				Unit 8	CA-0375	I/63472 Remote Relief Valve Position Indication - Unit 8	0.4916	21-Dec-18	
Fitness for service	2016	GIO-103	Implementation of Asset Management Activities	Bruce A & B	CA-0377	Implementation of Asset Management Activities for Safety Significant Assets	0.0570	31-Jan-18	
Fitness for service	2016	GIO-104	Ongoing Work on Bruce B Heat Transport Vibration Project	Bruce B	CA-0379	Bruce B Heat Transport Vibration Project	0.3460	22-Dec-17	NK21-CORR-00531-13357 / NK29-CORR-00531-13907

Appendix B – Regulatory Documents, Codes and Standards Considered for Assessment

Assessment Type:

NA: Not Assessed
CBC: Clause-by-Clause
PCBC: Partial Clause-by-Clause
CTC: Code-to-Code
HL: High Level
2SF: Assessment performed in another SFR
CV: Confirm Validity of Previous Assessments

The code effective date was August 31, 2014 for the Bruce A ISR and December 31, 2015 for the Bruce B PSR.

Document No.	Current Version	Title	Assessment Type															Bruce A	Bruce B
			SFR 1	SFR 2	SFR 3	SFR 4	SFR 5	SFR 6	SFR 7	SFR 8	SFR 9	SFR 10	SFR 11	SFR 12	SFR 13	SFR 14	SFR 15		
ANSI/HPS N13.1-1999	1999	Sampling and Monitoring Releases of Airborne Radioactive Substances From the Stacks and Duct of Nuclear Facilities														HL		X	X
ANSI/NIRMA CM 1.0-2007	2007	Guidelines for Configuration Management of Nuclear Facilities	HL							2SF								X	X
ASME BPVC Section III	2014 – BA 2015 – BB	Rules for Construction of Nuclear Power Plant Components	HL															X	X

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ASME BPVC Section VIII	2014 – BA 2015 – BB	Design and Fabrication of Pressure Vessels	HL															X	X
ASME B31.1	2014	Code for Power Piping	HL															X	X
CNSC EG-1	2005	Requirements and Guidelines for Written and Oral Certification Examinations for Shift Personnel at Nuclear Power Plants								NA		NA		NA				X	X
CNSC EG-2	2004	Requirements and Guidelines for Simulator-Based Certification Examinations for Shift Personnel at Nuclear Power Plants								NA		NA		NA				X	X
CNSC G-129	Rev 1 (2004/10)	Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'								2SF						HL	HL	X	X
CNSC G-144	2006/05	Trip Parameter Acceptance					HL											X	X
CNSC G-149	2000/10	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	HL				HL											X	X
CNSC G-228	2001/03	Developing and Using Action Levels								NA						HL	HL	X	X
CNSC G-276	2003/06	Human Factors Engineering Program Plans	2SF											HL				X	X
CNSC G-278	2003/06	Human Factors Verification and Validation Plan	NA											NA				X	X

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CNSC G-323	2007/08	Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement										2SF		NA				X	X
CNSC Internal Guidance 2009/05	2009	Requirements for the Requalification Testing of Certified Shift Personnel at Nuclear Power Plants								NA		NA		NA				X	X
CNSC Internal Guidance 2010/08	2010	CNSC Expectations for Licensee Hours of Work Limits - Objectives and Criteria								NA		NA		NA				X	X
CNSC P-325	2006/05	Nuclear Emergency Management													NA			X	X
CNSC R-10	1977/01	The Use of Two Shutdown Systems in Reactors	NA				NA			NA								X	X
CNSC R-77	1987/10	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	NA				CV		CV									X	X
CNSC R-116	1995/01	Requirements for Leak Testing Selected Sealed Radiation Sources															NA	X	X
CNSC RD/GD-99.3	2012/03	Public Information and Disclosure								NA	NA		NA		NA	NA		X	X
CNSC RD-204	2008/02	Certification of Persons Working at Nuclear Power Plants								NA		NA		NA				X	X
CNSC RD/GD-210	2012/11	Maintenance Programs for Nuclear Power Plants			NA	NA				NA		NA		NA				X	X

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CNSC RD-327	2010/12	Nuclear Criticality Safety	NA				NA												X
CNSC RD-346	2008/11	Site Evaluation for New Nuclear Power Plants	NA						CV							CV (NA)		X	X
CNSC RD-353	2008/11	Testing and Implementation of Emergency Measures													NA			X	
CNSC RD/GD-371	2011/11	Licence Application Guide: Nuclear Substances and Radiation Devices															NA	X	
CNSC REGDOC-1.6.1	2015/10	Licence Application Guide: Nuclear Substances and Radiation Devices															NA		X
CNSC REGDOC-2.10.1	2014/10	Nuclear Emergency Preparedness and Response													CBC			X	X
CNSC REGDOC-2.2.2	2014/08	Personnel Training								2SF		2SF		CBC				X	X
CNSC REGDOC-2.3.2	2014/10 – BA 2015/09 – BB	Accident Management Severe Accident Management Programs for Nuclear Reactors					PCBC								CBC			X	X
CNSC RD-360	2008/02	Life Extension Of Nuclear Power Plants	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	X	
CNSC REGDOC-2.3.3	2015/04	Periodic Safety Reviews	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA		X

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CNSC REGDOC-2.4.1	2014/05	Safety Analysis For Nuclear Power Plants (Current: Deterministic Safety Analysis)					CBC		PCBC									X	X
CNSC REGDOC-2.4.2	2014/05	Probabilistic Safety Assessment For Nuclear Power Plants	2SF					CBC										X	X
CNSC REGDOC-2.5.2	2014/05	Design of Reactor Facilities: Nuclear Power Plants	CBC			PCBC	PCBC	PCBC	PCBC					2SF				X	X
CNSC REGDOC-2.6.3	2014/03	Fitness for Service: Ageing Management				NA				NA								X	X
CNSC REGDOC-2.9.1	2013/09	Environmental Protection, Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills											2SF			CTC/HL		X	
	2013/09	Environmental Protection, Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills											NA			NA			X
CNSC REGDOC-3.1.1	2014/05	Reporting Requirements for Operating Nuclear Power Plants				NA				NA	NA	NA	NA			NA	NA	X	X
CNSC RD/GD-98	2012/06	Reliability programs for Nuclear Power Plants		NA		NA		NA										X	X
CNSC S-106	2006/05	Technical and Quality Assurance Requirements for Dosimetry Services															NA	X	X

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CSA B51	2014	Boiler, Pressure Vessel, and Pressure Piping Code	CV (NA)	NA														X	X
CSA N1600	2014	General Requirements for Nuclear Emergency Management Programs													HL			X	X
CSA N285.0	2012 Update 1 (2013/09)	General Requirements For Pressure-Retaining Systems And Components In CANDU Nuclear Power Plants	NA															X	
	2012 Update 1 (2013/09) Update 2 (2014/11)	General Requirements For Pressure-Retaining Systems And Components In CANDU Nuclear Power Plants	NA																X
CSA N285.4	2014	Periodic Inspection of CANDU Nuclear Power Plant Components		NA		HL												X	X
CSA N285.5	2013	Periodic Inspection of CANDU Nuclear Power Plant Containment Components				CTC/HL												X	X
CSA N285.8	2015	Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors				CV												X	

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	2015	Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors				HL													X
CSA N286.7-99	1999 (R2012)	Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants	NA	NA			NA	NA	NA									X	X
CSA N286-05	2012	Management System Requirements for Nuclear Power Plants	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	X	X
CSA N287.1	2014	General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	CTC/HL			CTC/CBC												X	X
CSA N287.2	2008 (R2013)	Material Requirements for Concrete Containment structures in CANDU nuclear power plants	CV (NA)															X	X
CSA N287.3	2014	Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	HL															X	X
CSA N287.4	2009 (R2014)	Construction, Fabrication, and installation requirements for Concrete Containment Structures for CANDU nuclear power plants	CV (NA)															X	X
CSA N287.5	2011	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	CV (NA)															X	X

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CSA N287.6	2011	Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants	NA																X
CSA N287.7	2008 with Update 1 (2010) (R2013)	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants				NA												X	X
CSA N288.1	2014	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities														HL		X	X
CSA N288.2	2014	Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors					NA								2SF			X	
	2014	Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors					HL								2SF				X
CSA N288.3.4	2013	Performance testing of nuclear air cleaning systems at nuclear facilities								2SF						HL		X	X
CSA N288.4	2010 – BA (R2015) – BB	Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	HL							NA						HL		X	X
CSA N288.5	2011	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills								NA						HL		X	X

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CSA N288.6	2012	Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills								NA						HL		X	X
CSA N288.7	2015	Groundwater protection programs at Class I nuclear facilities and uranium mines and mills														HL			X
CSA N289.1	2008 (R2013)	General requirements for seismic design and qualification of CANDU nuclear power plants	HL		HL													X	X
CSA N289.2	2010 – BA R2015 – BB	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	HL		HL													X	X
CSA N289.3	2010 – BA R2015 – BB	Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants	HL		HL													X	X
CSA N289.4	2012	Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants	HL		HL													X	X
CSA N289.5	2012	Seismic Instrumentation Requirements for CANDU Nuclear Power Plants	HL		HL													X	X
CSA N290.0-11	2011	General Requirements for safety systems of nuclear power plants	HL															X	X
CSA N290.1	2013	Requirements for the Shutdown Systems of CANDU Nuclear Power Plants	CBC				PCBC											X	X
CSA N290.2	2011	Requirements for emergency core cooling systems of nuclear plants	HL															X	X

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CSA N290.3	2011	Requirements for the containment system of nuclear plants	HL															X	X
CSA N290.4	2011	Requirements for Reactor Control Systems of Nuclear Power Plants	CV (NA)				CV											X	X
CSA N290.5	2006 (R2011)	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	CV (NA)				CV											X	X
CSA N290.6	2009 (R2014)	Requirements for monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	CV (NA)				CV											X	X
CSA N290.11	2013	Requirements for reactor heat removal capability during outage of nuclear power plants	HL															X	X
CSA N290.12	2014	Human Factors in Design for Nuclear Power Plants	2SF											CBC					X
CSA N290.13	2005 (R2010)	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	NA	NA	HL	NA	NA		CV									X	
	2005 (R2015)	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	NA	NA	NA	NA	NA		NA										X

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CSA N290.15	2010	Requirements for the safe operating envelope of nuclear power plants	NA	NA			NA	NA	NA	NA								X	X
CSA N291	2008 (R2013)	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	HL	HL		PCBC												X	
CSA N291	2015	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	HL	HL		PCBC													X
CSA N292.0	2014	General Principles for the Management of Radioactive Waste and Irradiated Fuel								NA									X
CSA N292.3	2014	Management of Low- and Intermediate-Level Radioactive Waste								2SF			CBC			2SF		X	X
CSA N293	2012	Fire Protection For CANDU Nuclear Power Plants	2SF						CTC/PCBC									X	X
CSA Z731	2003 (R2014)	General requirements for nuclear emergency management programs													NA			X	X
Darlington DG-38-03650-1	-	Purpose and Application of Nuclear Safety in Design	NA															X	X
Darlington DG-38-03650-2A	-	Common Mode Incidents – Overview and Design Requirements	NA						NA									X	X

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Darlington DG-38-03650-2B	-	Common Mode Incidents – Seismic Design	NA						NA									X	X
Darlington DG-38-03650-3	-	Limiting Consequential Damage of Postulated Pipe Ruptures	NA						NA									X	X
Darlington DG-38-03650-4	-	Shutdown Systems	NA															X	X
Darlington DG-38-03650-5	-	Emergency Coolant Injection	NA															X	X
Darlington DG-38-03650-6	-	Containment	NA															X	X
Darlington DG-38-03650-7	-	Extensions of the Containment Envelope	NA															X	X
Darlington DG-38-03650-8	-	Environmental Qualification of Safety Related Equipment	NA		NA													X	X
Darlington DG-38-03650-9	-	Safety Assessments	NA															X	X
IAEA GSR Part 7	2015	Preparedness and Response for a Nuclear or Radiological Emergency													HL				X
IAEA NS-G-3.2	2002/03	Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants														HL	NA	X	X

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IAEA NS-G-2.7	2002	Radiation Protection and Radioactive Waste Management in the Operation of Nuclear Power Plants															NA		X
IAEA RS-G-1.1	1999	Occupational Radiation Protection															NA		X
IAEA SSG-25	2013/03	Periodic Safety Review for Nuclear Power Plants	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	X	X
IAEA SSR-2/2	2011/07	Safety of Nuclear Power Plants: Commissioning and Operation Specific Safety Requirements										2SF	CBC					X	X
IAEA TECDOC-1141	2000	Operational Safety Performance Indicators for Nuclear Power Plants								NA									X
IAEA TECDOC-1338	2003	Configuration Management of Nuclear Power Plants								NA									X
INPO 91-014	Rev 1 (1995/10)	Guidelines for Radiological Protection at Nuclear Power Stations															NA	X	X
INPO 09-003	2016	Systematic Excellence in the Management of Design and Operating Margins								NA									X
INPO 05-008	2016	Radiological Protection at Nuclear Power Station								NA									X
INPO AP-913	2013	Equipment Reliability Process Description								NA									X

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INPO AP-928	2016	On-Line Work Management Process Description								NA									X
ISO 14001:2004	2004	International Standard Environmental Management Systems - Requirements								NA									X
NBCC	2010 First Revision and Errata (2012/12)	National Building Code of Canada	NA															X	
	2015	National Building Code of Canada	NA						CTC/HL										X
NFCC	2010	National Fire Code of Canada	NA															X	
	2015	National Fire Code of Canada	NA						CTC/HL										X
NFPA-805	2015	Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants	HL															X	X
NUREG-0700	2002	Human System Interface Design Review Guidelines												HL				X	X

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WANO GL 2004-01	2004	Guidelines for Radiological Protection at Nuclear Power Stations								2SF							CBC	X	X
WANO Good Practice ATL-11-006 Rev. 3	2011	Work Management Process Description								NA									X

Appendix C – **Global Assessment Framework**

C.1. Introduction

The objective of developing an assessment framework is to devise a systematic methodology and establish a common basis for assessing the relative importance of addressing Global Issues in terms of aspects such as their safety significance. The same framework is also used to assess the importance of practicable improvements and associated corrective actions for the development of the IIP.

The Global Assessment Framework (GAF) can be used for ranking and prioritization to answer questions such as the following:

- How should gaps be consolidated into GIOs?
- Which GIOs are the most important?
- How should the GIOs be addressed?
- Which GIOs should be addressed first?

These questions are interrelated, multi-facetted, and sometimes involve competing objectives. Moreover, the outcomes of potential answers to some of these questions are uncertain. An overarching set of values, principles, or goals is needed that can guide these activities and that would "drive" the whole process in a comprehensive, systematic and consistent manner through all the steps to develop an integrated and coordinated set of improvement initiatives.

More specifically, a process is needed to decide on the importance ranking and prioritization of the issues and potential improvements identified through the PSR and other assessment activities. This requires a multi-objective, multi-attribute decision support model to be formulated as follows:

- The multi-objective nature of the problem is described by decomposing overarching objectives into a hierarchical structure of sub-objectives called a value tree. The often conflicting nature of sub-objectives is accommodated through the allocation of relative weights to objectives attached to the same branch level of the value tree. Higher weights are assigned to branches for which enhancements provide the greatest benefit to safety, thereby risk-informing the value tree;
- A scoring system is devised that allows the decision maker to express preferences for resolving issues on a 5-point scale for each of two attributes: impact and time-to-take-effect. The impact score will take into account aspects such as contribution to defence-in-depth and safety significance, particularly impact on achieving safety goals;
- The impact and time scores are combined to produce an overall utility score for each issue that reflects a preference for resolutions that achieve high impact in a short time, but weigh impact somewhat higher in importance than time. Higher scores denote a greater preference for resolution, again risk-informing the process by placing priority on

issue resolution that will have the greatest value in supporting the underlying objective; and

- Finally, the value ranking of resolving an issue is calculated as the product of the relative weight of the corresponding objective and the utility score of the issue.

The resulting prioritization and ranking framework is embedded in the PSR database. The value tree will have three tiers below the cardinal objective. The first two tiers are utilized in the development, ranking and prioritization of Global Issues. The third tier is utilized in the development, ranking and prioritization of corrective actions to address Global Issues.

C.2. Structure of the Value Tree

The Value Tree described in this Appendix was developed by the Integrated Implementation Plan Project Team (IIPPT). The IIPPT is a multi-disciplinary team who has been involved in full application of the PSR with specific expertise in CANDU design, operation, inspection and maintenance, safety analysis, licensing, environmental issues, as well as management of the same.

C.3. Main Branches of the Value Tree

In structuring any decision problem it is important to determine exactly what the cardinal objective is and what sub-objectives need to be considered to support it in order to determine the fundamental dimensions of the values to make decisions. A useful technique to structure those values is to make use of a value tree. A value tree begins with a cardinal objective and a set of fundamental objectives as its main branches. Each fundamental objective is then expanded and supported with more specific objectives. A systematic comparison and assessment of these sub-objectives establishes how each is valued in achieving the cardinal objective.

C.3.1. Definition of Cardinal Objective

The two cardinal objectives in the long-term operation of Bruce A and Bruce B are well known and support Bruce Power's value of 'Safety First' and key result areas of Nuclear Performance Index (NPI), Safety Performance and Commercial Index. They are stated as follows:

- Enhanced confidence in the continued safety of Bruce A and Bruce B; and
- Enhanced confidence in the reliability of electricity production by Bruce A and Bruce B for an extended life.

Since these two cardinal objectives are mutually supportive and not in conflict, they can be combined into a single value statement, as follows:

- Enhanced confidence in the continued safety of Bruce A and Bruce B and reliability of electricity production for an extended life

This cardinal objective is also referred to as the Tier 0 Objective.

C.3.2. Main Branches of the Value Tree

Having defined "Enhanced confidence in the continued safety of Bruce A and Bruce B and reliability of electricity production for an extended life" as the cardinal objective of the Value Tree, the fundamental supporting objectives can be formulated in terms of the main branches of the Value Tree, also called "Tier 1 Branches", as shown in Figure 37.

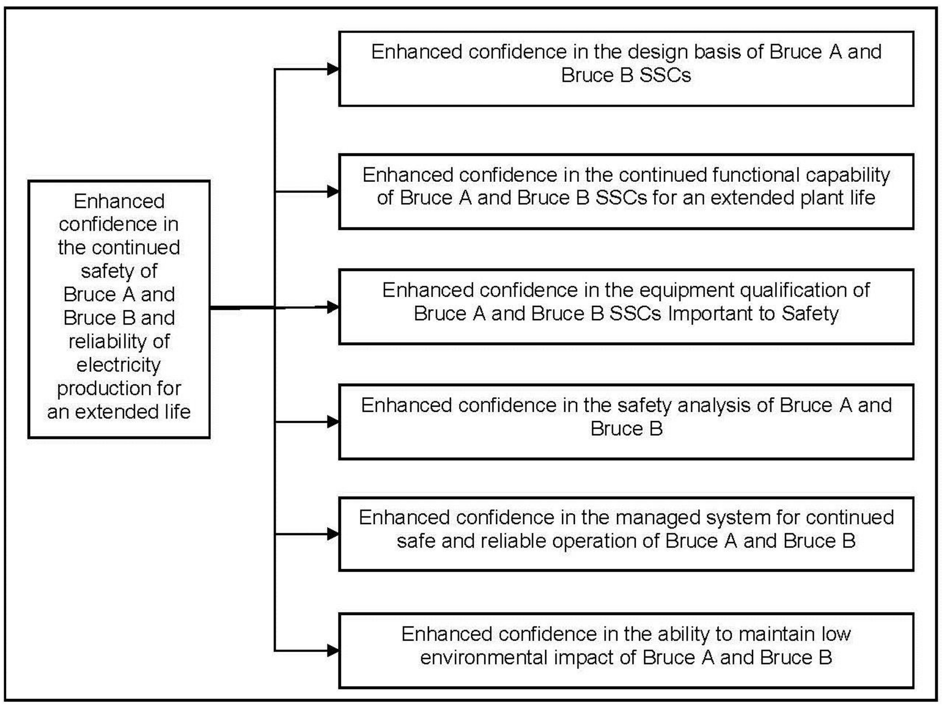

Figure 37: Tier 1 Branches

The basis for these branches is recognition that Bruce A and Bruce B, in compliance with its Power Reactor Operating Licence (PROL) and associated regulatory framework, must:

- Continue to conform with its design basis;
- Be operated well to achieve safety and reliable electricity production in accordance with its design in accordance with its managed system;
- Have an adequate safety and hazard analysis to demonstrate the facility's safety; and
- Achieve adequate environmental performance.

Meeting Canada's international obligations, i.e., safeguards, is not a branch of the Value Tree because safeguards do not contribute to the cardinal objective. This subject is treated outside of this methodology, as is security. Improvement opportunities and initiatives associated with safeguards and security are also excluded from the assessment. This means that any safeguards or security related improvement initiatives or commitments in-place will be implemented in accordance with Bruce Power's obligations under the law.

Since one of the key goals of the PSR is to assess whether enhancements should be made to better align with modern standards, these key elements are considered for further subdivision at the second level of the tree. It is important to note that no issue is advantaged or disadvantaged in terms of its rank by the extent of subdivision of the Value Tree. Each Tier 1 branch of the Value Tree is discussed below, together with an overall view of its current state that needs to be taken into consideration for pair wise comparisons.

C.3.2.1. Enhanced Confidence in the Design Basis of Bruce A and Bruce B SSCs

The design branch primarily considers maintenance of the current design basis and potential for improvements to the current design basis, as would be expected in modern standards given the age of Bruce A and Bruce B and considering the safety improvements implemented since the plant was put into operation.

The original design basis of Bruce A and Bruce B cannot meet all provisions of the applicable modern design codes and standards. In order to enhance the design basis of the current plant to a level comparable to that required for new NPPs, a significant level of design improvement based modern codes and standards applicable to a new NPP will be required. These improvements would require fundamental changes to the SSCs in place, most of which are impracticable to implement as they would require systemic changes affecting the plant SSCs as a whole and at times lead to conflicts amongst different requirements given the current configuration of the plant. However, since the beginning of Bruce A and Bruce B operation, safety upgrades and supplementary design and safety analyses have been continually implemented to comply with the those provisions of the PROL that required design upgrades with high priority and as an integral part of continued safe and reliable operation. A recent example is the implementation of practicable design changes in response to the Fukushima Action Items. It should also be noted that assessments of the design against modern versions of the original design requirements since the plant was put into operation have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth.

C.3.2.2. Enhanced Confidence in the Continued Functional Capability of Bruce A and Bruce B SSCs for an Extended Plant Life

This branch straddles design and operation, and is related to the confidence in the continued functional capability of Bruce A and Bruce B SSCs to meet their current design and operating requirements through monitoring, surveillance, testing, inspection and maintenance of SSCs in accordance with their design and operating envelope for an extended plant life. This branch is fundamentally different than the previous branch and deals with understanding of the current condition of Bruce A and Bruce B SSCs, ensuring their continued functional capability and maintaining them in this state. Functional capability of SCCs include structural integrity and operability within the prescribed safe operating envelope. The understanding of condition and ensuring functional capability of SSCs links the plant design basis and safe operation, as well as contributing to continued reliable electricity production for an extended life.

Confidence in the continued functional capability of Bruce A and Bruce B SSCs forms the basis of safe and reliable operation for an extended life. Given the age of Bruce A and Bruce B and the considerations for extended plant life, eliminating gaps and enhancing confidence in the understanding of the current condition of Bruce A and Bruce B SSCs, taking necessary actions to ensure their continued functional capability and maintaining them in this state continues to be the most important aspect of safe and reliable operation. Enhanced confidence in the continued functional capability of the as-built plant is the pillar of event-free operation.

C.3.2.3. Enhanced confidence in the equipment qualification of Bruce A and Bruce B SSCs Important to Safety

This branch straddles design and operation, and relates to equipment qualification, which invokes both design and programmatic elements. It is a separate Tier 1 branch because it was not always an explicit requirement originally considered in the design in a comprehensive and systematic manner as defined in modern codes and standards, at least not with respect to some internal and external hazards and certain accident conditions. This branch is also related to the safety analysis element as the robustness of the design due to initiating events is demonstrated via hazard and safety analysis.

The original design basis of Bruce A and Bruce B cannot not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design changes to meet modern codes and standards may be required to enhance the confidence in the design basis for equipment qualification as compared to a new NPP. Assessments performed to date have shown that the original design of Bruce A and Bruce B, including the safety upgrades (e.g., environmental and seismic qualification) incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth.

It should be noted that all three of the branches discussed so far address both plant safety and reliability considerations. The first two consider all Bruce A and Bruce B SSCs. The equipment qualification branch relates specifically to SSCs important to safety that are required to operate under accident conditions, as well as hazardous conditions due to internal or external events.

C.3.2.4. Enhanced confidence in the safety analysis of Bruce A and Bruce B

The safety analysis is maintained as a single branch in the first level of the Value Tree, primarily because the components of safety analysis (deterministic, probabilistic, hazard analysis) are all just different types of safety analysis that should fit within an overall integrated safety analysis framework. These components are recognized in lower tier branches.

A robust safety case relies on results of safety analyses that are executed in accordance with procedures that meet applicable quality assurance requirements, based on a systematic and comprehensive set of postulated initiating events, state-of-the-art analysis methodologies based on up-to-date experimental data and OPEX and input data that reflects actual plant. Since Bruce A and Bruce B was put into operation, hazard assessments and the scope of safety analysis have become a progressively more important area as a result of international OPEX and the resulting changes to requirements for systems important to safety, as well as development of the state-of-the-art analysis methodologies based on new experimental data and OPEX. Recently, the update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area.

C.3.2.5. Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B

The managed system branch is similarly kept to a single branch in the first level of the Value Tree. While there are many facets to the managed system that supports safe and reliable operation, they are all so closely inter-related that it would be necessary to include numerous branches to differentiate them at the level of the first branch of the Value Tree, which would result in unnecessary complexity at this level. Such differentiation is established more appropriately in Tier 2 and Tier 3 branches.

Bruce Power has a mature and robust managed system that is built around the principle of ‘safety first’ as the overarching objective. The managed system has built-in continuous improvement features. Bruce Power’s managed system meets all the regulatory requirements in the PROL associated with management of the plant operations.

C.3.2.6. Enhanced confidence in the ability to maintain low environmental impact of Bruce A and Bruce B

The environmental branch is also kept to a single branch in the first level. Differentiation between normal, accident and post-accident conditions takes place at the Tier 2 branch.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of ‘safety first’ principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents and meet CNSC’s expectations as a high priority.

C.4. Expanding the Value Tree Structure – Tier 2 and Tier 3 Branches

Each of the Tier 1 objectives was expanded into specific supporting objectives at Tier 2, and similarly each Tier 2 objective was expanded into more specific supporting objectives at Tier 3. The resulting Value Tree is shown Table 37 in full, together with a description of each objective.

Safety Factors have been mapped to the Value Tree as shown in the second last column of Table 37. This mapping demonstrates that all micro-gaps and findings identified through the

PSR process and the associated improvement opportunities can be evaluated in terms of priority and ranking within the framework of the Value Tree as part of the Global Assessment and Integrated Implementation Plan development.

The last column of Table 37 indicates those levels of defence-in-depth supported by each of the Tier 1, Tier 2 and Tier 3 sub-objectives.

Table 37: Expanded Value Tree Objectives

T1 #	Tier 1 Contribution to Cardinal Values Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 Tier 3 Description	Applicable SFs	DID Level at Tier 2
1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1.1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1 & 3	1,2,3
		1.2	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	1.2.1	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	1, 2, 3 & 4	1,2,3
		1.3	Enhanced confidence that the design of SSCs meets modern standards	1.3.1	Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards	1 & 3	1,2,3
				1.3.2	Enhanced confidence that the design specification/ analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards	1 & 3	
2	Enhanced confidence in the continued functional capability of Bruce A and Bruce B SSCs for an extended plant life	2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).	2.1.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) for an extended plant life	2 & 4	1,2,3,4
				2.1.2	Enhanced confidence in knowledge about the current condition of SSCs Important to Reliability (SIR) for an extended plant life	2 & 4	
		2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment	2.2.1	Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life	2 & 4	1,2,3,4
				2.2.2	Enhanced confidence in restoring SIR to a state that achieves the intended functionality and extended plant life	2 & 4	

T1 #	Tier 1 Contribution to Cardinal Values / Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 / Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 / Tier 3 Description	Applicable SFs	DID Level at Tier 2
			Reliability Program.				
		2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life. This is achieved through the Maintenance Program activities.	2.3.1	Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life	2 & 4	1,2,3,4
				2.3.2	Enhanced confidence in maintaining SIR in a state that achieves reliable operation and safety performance and extended plant life	2 & 4	
		2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.	2.4.1	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life	2 & 4	1,2,3,4
3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety	3.1	Enhanced confidence in the design specification and implementation of equipment qualification conditions for Systems Important to Safety	3.1.1	Enhanced confidence in the current environmental qualification requirements of SIS resulting from deterministic safety analysis of Design Basis Accidents (DBAs)	1, 3 & 5	1,2,3,4
				3.1.2	Enhanced confidence in the equipment qualification requirements resulting from hazards analysis of internal and external events	1, 3, 6 & 7	
				3.1.3	Enhanced confidence in the equipment qualification conditions and requirements for SSCs not already specified in safety analysis or hazards analysis (e.g. Severe Accident (SA) conditions)	1, 3, 6 & 7	1,2,3,4
4	Enhanced confidence in the safety analysis of Bruce A and Bruce B	4.1	Enhanced confidence in the comprehensiveness of the safety analysis	4.1.1	Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record	5, 6 & 7	1,2,3,4
				4.1.2	Enhanced confidence in the definition of initiating events and combinations thereof in the current analysis of record	5, 6 & 7	
				4.1.3	Enhanced confidence in the coverage of all initiating events and combinations thereof of the current safety analysis of record	5, 6 & 7	
		4.2	Enhanced confidence in conformance with the applicable safety analysis methods and associated acceptance criteria	4.2.1	Enhanced confidence in the degree to which software used for accident analysis has been validated	5, 6 & 7	1,2,3,4
				4.2.2	Enhanced confidence in the degree to which acceptance criteria used in safety	5, 6 & 7	

T1 #	Tier 1 Contribution to Cardinal Values Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 Tier 3 Description	Applicable SFs	DID Level at Tier 2
					analysis is supported by experimental or operational data		
				4.2.3	Enhanced confidence in the application of modern methodologies and criteria in the conduct of safety analysis	5, 6 & 7	
5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B	5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience	5.1.1	Enhanced confidence in the selection and training of staff	10 & 12	1,2,3,4
				5.1.2	Enhanced confidence in the dissemination and assimilation of internal and external operating experience	10 & 12	
		5.2	Enhanced confidence in the effectiveness of technical and administrative documentation and interfaces for operators, maintainers and operations support staff	5.2.1	Enhanced confidence in the comprehensiveness and effectiveness of procedures	11	1,2,3,4 ,5
				5.2.2	Enhanced confidence in the appropriateness, validity and timeliness of plant and process information	1, 2, 3, 5, 6, 10, 11 & 12	
				5.2.3	Enhanced confidence in the appropriateness of plant control interfaces (human factors)	12	
		5.3	Enhanced confidence in a safe work environment	5.3.1	Enhanced confidence in radiation protection	15	1,2,3,4 ,5
				5.3.2	Enhanced confidence in conventional health and safety	8	
		5.4	Enhanced confidence in organizational effectiveness by establishing a management structure, processes and supporting infrastructure that can enable continuous improvement of safe plant operation and safety culture	5.4.1	Enhanced confidence in management system structure processes and supporting infrastructure	10 & 11	1,2,3,4 ,5
				5.4.2	Enhanced confidence in safety culture	10, 11 & 12	
				5.4.3	Enhanced confidence in performance monitoring and corrective action	8, 9, 10, 11 & 12	
6	Enhanced confidence in the ability to maintain low environmental impact of Bruce A and Bruce B	6.1	Enhanced confidence in maintaining a low environmental impact during normal operations	6.1.1	Enhanced confidence in low impact of radioactive releases	14 & 15	1,2,3,4 ,5
				6.1.2	Enhanced confidence in low impact of non-radiological releases	14	
		6.2	Enhanced confidence in the ability to mitigate releases associated with external/internal events	6.2.1	Enhanced confidence in the ability to mitigate releases associated with anticipated operational occurrences and design basis events	13 & 14	1,2,3,4 ,5
				6.2.2	Enhanced confidence in the ability to mitigate releases associated with beyond design basis events	13	

C.5. Validation of Value Tree Structure

The Value Tree structure was further validated by comparison to the following similar hierarchical structures:

- The CNSC Safety and Control Areas (REGDOC-2.3.3 Periodic Safety Reviews, Appendix B);
- World Association of Nuclear Operators (WANO) Performance Objectives and Criteria, "WANO Performance Objectives and Criteria", World Association of Nuclear Operators, Revision 3, January 2005);
- Institute of Nuclear Power Operations (INPO) Performance Objectives and Criteria "Performance Objectives and Criteria", (Institute of Nuclear Power Operations, INPO 05-003, May 2005).

These structures were also related to the PSR Safety Factor topics to find a common basis for comparison. Before comparing the Value Tree to these other hierarchical systems, it should be noted that these hierarchies simply provide a useful framework for the respective organizations that created them to perform their work of evaluating nuclear plant safety and/or their operational effectiveness. Bruce Power has adapted many of the best practices used or recommended by these organizations. These frameworks can therefore be viewed as work breakdown structures for safe and reliable operation. As such, these structures, on their own, do not provide any sense of prioritization or ranking of their constituent elements. In contrast, the Value Tree was deductively derived from a single objective to provide a decision support tool to assist in making decisions relating to a specific nuclear plant, in a specific state, at a very specific time in its life-cycle.

Since value trees are constructed to support specific decisions, they should only contain elements that are germane to the decision at hand and are not expected to be comprehensive in the sense of addressing all potential areas of interest. In fact, a common mistake in constructing value trees is to make them too complicated. The number of fundamental objectives should be comprehensive enough to address the important dimensions of the decision. Including extraneous objectives as some of the main branches of the Value Tree for the sake of "completeness" only serves to dilute the weight of the really fundamental objectives and may result in a decision tool that lacks any real discriminatory power.

The review in Table 38 showed that the CNSC Safety and Control Areas cover all the PSR Safety Factors comprehensively, as well as adding three additional Control Areas; Security, Safeguards and Non-Proliferation and Packaging and Transport, which is not covered among the PSR Safety Factors but addressed separately. The WANO and INPO and Performance Objectives and Criteria are very similar. Their coverage of the PSR Safety Factors shows an emphasis on programs with only minor attention being paid to the aspects covered under SF-1 to SF-4. The safety analysis Safety Factors SF-5, SF-6, and SF-7 are not covered at all. This was not unexpected since the WANO and INPO focus is primarily on excellence of operating and support practices and not on plant design and condition.

As demonstrated in Table 38, there is a good fit between the Safety Factors covered by the Value Tree and those covered by the CNSC Safety and Control Areas except for the previously noted exceptions of Security, Safeguards and Non-Proliferation and Packaging and Transport that are dealt with outside the PSR.

In addition to comparing the Value Tree against the CNSC, WANO, and INPO structures as outlined above, it was also calibrated for completeness against the 5 levels of defence-in-depth, defined in INSAG-10 as shown in Table 39. The results of this mapping are included in Table 37 and show that the Value Tree covers the levels of defence-in-depth comprehensively.

Table 38: Review of CNSC Safety Control Areas against PSR Safety Factors and Value Tree Tier 2 Objectives

CNSC Safety and Control Area		Relevant Safety Factors	Relevant Tier 2 Objectives
Title	Description		
Management system	The framework that establishes the processes and programs required to ensure an organization achieves its safety objectives, continuously monitors its performance against these objectives, and fosters a healthy safety culture.	SF-8 Safety Performance SF10-Organization and Administration	5.2 5.3 5.4
Human performance management	The activities that enable effective human performance through the development and implementation of processes that ensure a sufficient number of licensee personnel are in all relevant job areas and have the necessary knowledge, skills, procedures and tools in place to safely carry out their duties.	SF-9 Use of Experience from other Plants and Research Findings SF-10 Organization and Administration SF-11 Procedures SF-12 Human Factors	5.1 5.2 5.4
Operating performance	This includes an overall review of the conduct of the licensed activities and the activities that enable effective performance.	SF10-Organization and Administration SF-8 Safety Performance SF-9 Use of Experience from other Plants and Research Findings	5.1 5.2 5.3 5.4
Safety analysis	Maintenance of the safety analysis that supports the overall safety case for the facility. Safety analysis is a systematic evaluation of the potential hazards associated with the conduct of a proposed activity or facility and considers the effectiveness of preventative measures and strategies in reducing the effects of such hazards.	SF-5 Deterministic Safety Analysis SF-6 Probabilistic Safety Analysis SF-7 Hazard Analysis	4.1 4.2
Physical design	The activities that affect the ability of structures, systems and components to meet and maintain their design basis given new information arising over time and taking changes in the external environment into account.	SF-1 Plant Design SF-2 Actual Condition of Systems, Structures, and Components SF-3 Equipment Qualification SF-4 Ageing	1.1 1.2 1.3 2.1 3.1
Fitness for service	The activities that affect the physical condition of structures, systems and components to ensure that they remain effective over time. This area includes programs that ensure all equipment is available to perform its intended design function when called upon to do so.	SF-1 Plant Design SF-2 Actual Condition of Systems, Structures, and Components SF-3 Equipment Qualification SF-4 Ageing	2.1 2.2 2.3 2.4

CNSC Safety and Control Area		Relevant Safety Factors	Relevant Tier 2 Objectives
Title	**Description**		
Radiation protection	The implementation of a radiation protection program in accordance with the Radiation Protection Regulations. This program must ensure that contamination levels and radiation doses received by individuals are monitored and controlled, and maintained as low as reasonably achievable (ALARA)	SF-8 Safety Performance SF-15 Radiation Protection measures implemented in the plant design	5.3
Conventional health and safety	The implementation of a program to manage workplace safety hazards and to protect personnel and equipment.	SF-8 Safety Performance SF10-Organization and Administration SF-11 Procedures	5.3
Environmental protection	The programs that identify, control and monitor all releases of radioactive and hazardous substances and effects on the environment from facilities or as the result of licensed activities.	SF-14 Radiological Impact on the Environment SF-8 Safety Performance	6.1
Emergency management and fire protection	The emergency plans and emergency preparedness programs which exist for emergencies and for non-routine conditions. This area also includes any results of participation in exercises.	SF-13 Emergency Planning	6.2 3.1
Waste management	The internal waste-related programs that form part of the facility's operations up to the point where the waste is removed from the facility to a separate waste management facility. This area also covers the planning for decommissioning.	SF10-Organization and Administration SF-15 Radiation Protection measures implemented in the plant design	6.1 5.3
Security	The programs required to implement and support the security requirements stipulated in the regulations, in the license, in orders, or in expectations for the facility or activity.	Addressed separately	NA
Safeguards and non-proliferation	The programs required for the successful implementation of the obligations arising from the Canada/IAEA safeguards agreements as well as all other measures arising from the Treaty on the Non-Proliferation of Nuclear Weapons.	Addressed separately	NA
Packaging and transport	The programs that manage the safe packaging and transport of nuclear substances and radiation devices to and from the licensed facility.	Addressed separately	NA

Table 39: INSAG-10 Levels of Defence-in-Depth

Levels of DID (INSAG-10)	Objective	Essential means
1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
2	Control of abnormal operation and detection of failures	Control, limiting and protection systems and other surveillance features
3	Control of accidents within the design basis	Engineered safety features and accident procedures
4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

C.6. Assigning Weights to Objectives in the Value Tree

To account for differences in importance between the objectives that comprise the Value Tree, relative weights were assigned to each objective. The method used for assigning the weights was based on the well known AHP ("The Analytic Hierarchy Process", T.L. Saaty, McGraw Hill Inc, 1980) and comprised the following:

- Use of pair-wise comparisons to rank all of the objectives attached to the same branch in terms of importance on a scale from 1 to 9. If there are 'n' objectives attached to the same branch, the result is an n x n reciprocal matrix.
- Computing the eigenvalues of the matrix and find the eigenvector corresponding to the largest eigenvalue λ_{max}.
- Normalizing the largest eigenvector and decompose into its n components.
- Assigning the components of the normalized eigenvector as weights to the corresponding objectives on the Value Tree.
- Computing the consistency index (CI) as $CI = (\lambda_{max} - n) / (n – 1)$
- Computing the Consistency Ratio (CR), as the ratio of the Consistency Index (CI) for a particular set of judgments, to the random index (RI) for a matrix of the same size as published in ("The Analytic Hierarchy Process", T.L. Saaty, McGraw Hill Inc, 1980). If CR is less than 10%, the judgment results are considered acceptable.

In practice, this process was approximated by calculating the average of the row entries of the reciprocal matrix after the columns were normalized. The PSR Database is an integrated tool based on Microsoft Access technology that provides traceability of all corrective actions to individual CARDs through groupings into GIOs down to the original source of the issue, i.e., micro-gaps or initiatives considered relevant. The Value Tree and weighting system is also built into the tool and the tool takes care of all ranking calculations. It is common practice in tools of this kind to ease the calculation burden by approximating the calculation of the normalized components of the eigenvector associated with the largest eigenvalue of the reciprocal matrix value by the average of the rows of the normalized columns. For reciprocal matrices up to size 3 x 3, this is considered to be very accurate. However, a 6 x 6 matrix is required to determine the Tier 1 weights, and thus the difference for this case was quantified, as this provides an indication of the worst level of accuracy. The results are shown in Table 40.

Table 40: Evaluation of Accuracy of the Method to Approximate Eigenvalue Calculations

T1 Objective	Averaging Method Used in PSR Database (A)	Eigenvalue Method (B)	Difference (A-B)	% Difference [(A-B)/B]*100 (Absolute Value)
1	0.0350	0.0336	0.0014	4.0414
2	0.4916	0.5057	-0.0141	2.7798
3	0.1475	0.1438	0.0037	2.5925
4	0.1401	0.1369	0.0032	2.3320
5	0.0457	0.0431	0.0026	5.9926
6	0.1401	0.1369	0.0032	2.3320
Average Percentage Difference				3.345

The average value of the percentage difference is 3.3% between the two values, which is well within the acceptable consistency ratio of 10%. Therefore, the method used in the tool is considered sufficiently accurate.

To compensate for the fact that not all branches of the Value Tree have the same number of tiers, the PSR database also performs a second normalization of the all weights to make them truly relative and comparable. The results of the weight assignment process are presented in Table 41. The rationale used to determine the weights is presented in Section C.10.

Table 41: Assignment of Weights to Objectives in the Value Tree

Tier 1			Tier 2				Tier 3			
#	Tier 1 Description	Tier 1 Weight *(A)*	#	Tier 2 Description	Tier 2 Weight *(B)*	Ideal Mode Branch Weight (Note 1)	#	Tier 3 Description	Tier 3 Weight (C)	Ideal Mode Branch Weight (Note 2)
1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	0.0350	1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	0.4615	0.0350	1.1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1.0000	0.0350
			1.2	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	0.4615	0.0350	1.2.1	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	1.0000	0.0350
			1.3	Enhanced confidence that the design of SSCs meets modern standards	0.0769	0.0058	1.3.1	Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards	0.1250	0.0008
							1.3.2	Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards	0.8750	0.0058
2	Enhanced confidence in the continued functional	0.4916	2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to	0.0559	0.0570	2.1.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) for an extended plant life	0.8750	0.0570

Tier 1			Tier 2				Tier 3			
#	Tier 1 Description	Tier 1 Weight *(A)*	#	Tier 2 Description	Tier 2 Weight *(B)*	Ideal Mode Branch Weight (Note 1)	#	Tier 3 Description	Tier 3 Weight (C)	Ideal Mode Branch Weight (Note 2)
	capability of Bruce A and Bruce B SSCs for an extended plant life			Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).			2.1.2	Enhanced confidence in knowledge about the current condition of SSCs Important to Reliability (SIR) for an extended plant life	0.1250	0.0081
			2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment Reliability Program.	0.3158	0.3216	2.2.1	Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life	0.8750	0.3216
							2.2.2	Enhanced confidence in restoring SIR to a state that achieves the intended functionality and extended plant life	0.1250	0.0459
			2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life. This is achieved through the Maintenance Program activities.	0.4827	0.4916	2.3.1	Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life	0.8750	0.4916
							2.3.2	Enhanced confidence in maintaining SIR in a state that achieves reliable operation and safety performance and extended plant life	0.1250	0.0702
			2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.	0.1455	0.1482	2.4.1	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life	1.0000	0.1482

Tier 1			Tier 2				Tier 3			
#	Tier 1 Description	Tier 1 Weight *(A)*	#	Tier 2 Description	Tier 2 Weight *(B)*	Ideal Mode Branch Weight (Note 1)	#	Tier 3 Description	Tier 3 Weight (C)	Ideal Mode Branch Weight (Note 2)
3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety	0.1475	3.1	Enhanced confidence in the design specification and implementation of equipment qualification conditions for Systems Important to Safety	1.0000	0.1475	3.1.1	Enhanced confidence in the current environmental qualification requirements of SIS resulting from deterministic safety analysis of Design Basis Accidents (DBAs)	0.1667	0.0295
							3.1.2	Enhanced confidence in the equipment qualification requirements resulting from hazards analysis of internal and external events	0.8333	0.1475
4	Enhanced confidence in the safety analysis of Bruce A and Bruce B	0.1401	4.1	Enhanced confidence in the comprehensiveness of the safety analysis	0.8000	0.1401	4.1.1	Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record	0.7014	0.1401
							4.1.2	Enhanced confidence in the definition of initiating events and combinations thereof in the current analysis of record	0.2132	0.0426
							4.1.3	Enhanced confidence in the coverage of all initiating events and combinations thereof of the current safety analysis of record	0.0853	0.0170
			4.2	Enhanced confidence in conformance with the applicable safety analysis methods and associated acceptance criteria	0.2000	0.0350	4.2.1	Enhanced confidence in the degree to which software used for accident analysis has been validated	0.2267	0.0113
							4.2.2	Enhanced confidence in the degree to which acceptance criteria used in safety analysis is supported by experimental or operational data	0.7015	0.0350
							4.2.3	Enhanced confidence in the application of modern methodologies and criteria in the conduct of safety analysis	0.0718	0.0036

Tier 1			Tier 2				Tier 3			
#	Tier 1 Description	Tier 1 Weight *(A)*	#	Tier 2 Description	Tier 2 Weight *(B)*	Ideal Mode Branch Weight (Note 1)	#	Tier 3 Description	Tier 3 Weight (C)	Ideal Mode Branch Weight (Note 2)
5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B	0.0457	5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience	0.0745	0.0087	5.1.1	Enhanced confidence in the selection and training of staff	0.7500	0.0087
							5.1.2	Enhanced confidence in the dissemination and assimilation of internal and external operating experience	0.2500	0.0029
			5.2	Enhanced confidence in the effectiveness of technical and administrative documentation and interfaces for operators, maintainers and operations support staff	0.3939	0.0457	5.2.1	Enhanced confidence in the comprehensiveness and effectiveness of procedures	0.0738	0.0052
							5.2.2	Enhanced confidence in the appropriateness, validity and timeliness of plant and process information	0.2828	0.0201
							5.2.3	Enhanced confidence in the appropriateness of plant control interfaces (human factors)	0.6434	0.0457
			5.3	Enhanced confidence in a safe work environment	0.3747	0.0435	5.3.1	Enhanced confidence in radiation protection	0.7500	0.0435
							5.3.2	Enhanced confidence in conventional health and safety	0.2500	0.0145
			5.4	Enhanced confidence in organizational effectiveness by establishing a management structure, processes and supporting infrastructure that can enable continuous improvement of safe plant operation and safety culture	0.1569	0.0182	5.4.1	Enhanced confidence in management system structure, processes and supporting infrastructure	0.0796	0.0022
							5.4.2	Enhanced confidence in safety culture	0.2648	0.0074
							5.4.3	Enhanced confidence in performance monitoring and corrective action	0.6555	0.0182

Tier 1			Tier 2				Tier 3			
#	**Tier 1 Description**	**Tier 1 Weight *(A)***	**#**	**Tier 2 Description**	**Tier 2 Weight *(B)***	**Ideal Mode Branch Weight (Note 1)**	**#**	**Tier 3 Description**	**Tier 3 Weight (C)**	**Ideal Mode Branch Weight (Note 2)**
6	Enhanced confidence in the ability to maintain low environmental impact of Bruce A and Bruce B	0.1401	6.1	Enhanced confidence in maintaining a low environmental impact during normal operations	0.8333	0.1401	6.1.1	Enhanced confidence in low impact of radioactive releases	0.8750	0.1401
							6.1.2	Enhanced confidence in low impact of non-radiological releases	0.1250	0.0200
			6.2	Enhanced confidence in the ability to mitigate releases associated with external/internal events	0.1667	0.0280	6.2.1	Enhanced confidence in the ability to mitigate releases associated with anticipated operational occurrences and design basis events	0.8333	0.0280
							6.2.2	Enhanced confidence in the ability to mitigate releases associated with beyond design basis events	0.1667	0.0056

Notes:

1. Ideal Mode Branch Weight=$(A \times B) / B_{max}$
2. Ideal Mode Branch Weight=$(((A \times B) / B_{max}) \times C) / C_{max}$

C.7. Expressing Preferences

The second step of developing a complete decision support system was the development of a measure or rate to judge the relative impact of resolving an issue on the objective that it is associated with. A preference rating system using two attributes is used:

- The **Time** attribute measures the impact of resolving an issue by answering the following question: “If one could somehow correct Issue X immediately (i.e., the corrective action happens overnight), how long would it take to see the assessed micro-gap disappear in relation to the corresponding objective?”
- The **Impact** attribute measures how directly or strongly the issue impacts the objective by asking: “If one could somehow correct Issue X immediately (i.e., the corrective action happens overnight), how direct or big would the impact be on the improvement in the objective?”

For both attributes a rating system on a scale of 1 to 5 was developed as shown in Table 42 and Table 43.

Table 42: Scoring System for the Time Attribute

Time Rating	Definition
1	Resolving the issue will take at least 10 years to have its effect on the objective
2	Resolving the issue will take at least 8 to 10 years to have its effect on the objective
3	Resolving the issue will take at least 6 to 10 years to have its effect on the objective
4	Resolving the issue will take at least 4 to 6 years to have its effect on the objective
5	Resolving the issue will take up to 3 years to have its effect on the objective

Table 43: Scoring System for the Impact Attribute

Impact Rating	Definition
1	Resolving the issue will have an indirect and negligible impact on the objective
2	Resolving the issue will have an indirect and minor impact on the objective
3	Resolving the issue will have an direct and minor impact on the objective
4	Resolving the issue will have an indirect and major impact on the objective
5	Resolving the issue will have an direct and major impact on the objective

To assist assessors in assigning Impact ratings, the terms in Table 43 are defined as indicated in the matrix shown in Table 44. This matrix also relates the INSAG-10 levels of defence-in-depth and the nature of corrective actions needed to resolve an issue to the type of Impact.

The resolution attributes (i.e., Direct, Indirect, Major, Minor) were considered at the level of the Value Tree at which the issue is mapped and within the context of the four rows of Table 44:

1. Maintaining or Improving the Design Basis (physical changes to SSCs);
2. Improving Safety or Operational Performance;
3. Reduction of Uncertainty in the design basis and operation of the Physical Plant and its Operation; and
4. Improvement of the Managed System (enablers) to achieve safe operation and reliable electricity production.

Each improvement is also evaluated within the context of four columns in Table 44 to establish their contribution as barriers to achieve safe and reliable, i.e., event free operation:

1. New barriers and practices;
2. Augmentation (Recovery) of the Current Barriers and Practices;
3. Improvement (Effectiveness) of the Current Barriers and Practices; and
4. Modernization of Current Barriers and Practices.

This approach ensures that all issues/initiatives are considered deterministically and in terms of their contribution to overall risk reduction associated with plant operation.

The matrix shown in Table 45 provides a numerical value to be selected in Table 44 that corresponds to major/minor, direct/indirect aspects of the impact utility assessment. The matrix comprises four types of improvements and four categories of contribution to establishing barriers for event-free operation.

For example, within this matrix, in relative terms, any improvement activity that results in achieving event free operation through physical improvements to the plant through new or augmented barriers and practices has the most “direct” and “major” impact on the cardinal objective of the Value Tree (safe plant operation and electricity production reliability). On the other hand, in relative terms, modernization of a current effective barrier or practice to an enabler could have “indirect” and “minor” impact on the cardinal objective of the Value Tree.

Note that expressions of preferences are determined throughout by viewing initiatives in isolation without initially taking potential synergies or prerequisites into account. This was intentional to make prioritization tractable and to clearly identify the preference without subconsciously taking feasibility into account. Interdependence, synergy and feasibility are taken into account in the scoping and scheduling of corrective actions in the IIP.

The evaluation of each issue or initiative is conducted in two steps:

1. In terms of its best fit considering its contribution to defence-in-depth and improvement of barriers and practices assigning the value in the corresponding box.

2. If the contribution to any of the Bruce Power safety goals is judged to be significant (e.g. by an order of magnitude) then the value obtained in step one is automatically changed to 5.

The following should be considered in the use of Table 44:

Values in boxes represent the Impact-Utility Score.

- **New** means there is no barrier or practice in place.
- **Augmentation** means current barrier is not complete or requires recovery or execution gaps in current practices or the modern codes and standards have complementary requirements that are not in place.
- **Improvement** means current barriers or practices are not kept fully effective.
- **Modernization** means current barriers or practices are effective but documentation and/or practices need to be updated to reflect current trends, state of the art approaches, terminology, etc., with no impact on operational performance.

Table 44: Impact Evaluation Matrix

INSAG-10 Level of DID	Row	Type of Issue/Corrective Action Requirement		Column: Nature of Improvement in Terms of Barriers and Practices			
				1	2	3	4
				New Barriers and Practices	Augmentation (Recovery) of Current Barriers and Practices	Improvement (Effectiveness) of Current Barriers and Practices	Modernization of Current Barriers and Practices
1	**1**	Maintain or Improve Design Basis (Includes Modifications or Replacements) & Implementation		5	5	4	2
1 & 2	**2**	Improve Safety or Operational Performance – (Plant Monitoring, Testing, Inspection & Maintenance, Configuration Management, Prevention of or Response to Events)		5	4	3	2
3 & 4	**3**	Reduce Uncertainty in design basis, engineered safety features and improve engineering analysis and procedures– (e.g. Engineering reviews, studies, analysis, FFS, LCM, AMP, SOE)		4	3	2	1
1-5	**4**	Improve Managed System and Organizational Effectiveness (Process, program, procedure)	a. Field Impact (e.g., operating, outage, maintenance, field procedures)	4	3	2	1
1-5			b. Managed system and support processes Impact (e.g., general training, OPEX)	3	2	2	1

C.8. Quantifying the Utility of Issue Resolution

The time and impact attributes discussed in Section C.7 express the preferences of the decision maker but not uncertainty. To quantify the utility of resolving an issue, it is necessary to combine the time and impact attributes to obtain a numerical value that represents the utility of the two parameters each rated on their 1 to 5 scale. This is achieved through the use of utility functions such as the following:

$U(x) = 1 - e^{-x/R}$

where:

- U is the utility of attribute x, with 0 < U < 1
- x is the score of the attribute on some scale (in this case, the 1 to 5 scale)
- R is an adjustable parameter that expresses preference; in this case, by adjusting R the utility is more strongly weighted towards solutions with high Impact or Time, versus simply the linear relationship provided by a 1 to 5 scale

Since the decision model expresses preference in terms of two attributes (Time and Impact) a two parameter utility function is formulated. Given utility independence, the following function is used:

$U(i,t) = k_i\ U_I(i) + k_t\ U_T(t) + (1\text{-}k_i\text{-}k_t)\ U_I(i)U_T(t)$

Where:

- U(i,t) is the utility of resolving the Issue, taking into account the Time and Impact scores each assessed on a 1 to 5 scale
- k_i is the contribution of the Impact attribute
- $U_I = 1 – e^{–i/R}{}_I$
- k_t is the contribution of the Time attribute
- $U_T = 1 – e^{–t/R}{}_T$

To use the two–parameter utility function, the IIPPT evaluated its preferences for Impact (R_I) and Time (R_T), as well as k_i = U(5,1) and k_t = U(1,5). This evaluation resulted in the following values:

- $k_i = 0.3$
- $R_I = -5.0$
- $k_t = 0.1$
- $R_T = -2.0$

The result is the utility matrix given in Table 45. To use the matrix, the Time and Impact values are determined on a 1 to 5 scale, and the resulting utility (score) associated with resolving the Issue is given by the number in the cell associated with the Time and Impact scores.

Table 45: Utility Matrix

		Time				
		1	**2**	**3**	**4**	**5**
Impact	**1**	0.00	0.01	0.03	0.05	0.10
	2	0.05	0.08	0.11	0.17	0.26
	3	0.12	0.15	0.21	0.31	0.46
	4	0.20	0.25	0.34	0.48	0.70
	5	0.30	0.37	0.49	0.68	1.00

The table indicates that the IIPPT strongly prefers solutions that:

- Have greater impact on the objective in shorter time, i.e., (4,4) is 6 times more preferable than (2,2)
- Have larger impact versus one that can be done quickly but with little impact, i.e., (1,5) is three times more preferable than (5,1)

C.9. Using the Decision Model

C.9.1. Application of the Decision Model to the Global Assessment Methodology

Since the Value Tree provides an integrated, coherent set of values for the assessment of the Global Issues and the IIP, it is used to guide the following processes:

1. Initial Ranking of GIs and GIOs against the second tier of the Value Tree; and
2. Preliminary and Final Ranking of CAs against the third tier of the Value Tree.

C.10. Pair-wise Comparison of the Value Tree Elements

C.10.1. Introduction

The overall objective of the Bruce A and Bruce B PSR is to obtain a practicable set of improvements that can be implemented over the PSR period and during the planned MCR outages in Units 3 to 8, to support ongoing operation enhancing safety and reliability to support long term operation. The time horizon that establishes the scope of the MCR is 30 or more years of safe and reliable operation. Hence, the PSR will cover a 10-year period, as there is an expectation that a PSR will be performed on approximately a 10-year cycle, as well as assuring that Units 3 to 8 can be operated well beyond a typical 10-year PSR cycle.

C.10.2. Description of the Process and Results

This Appendix describes the rationale behind the results of the assigned relative weights to the objectives that make up the Value Tree that forms the backbone of the decision support model. The weights will be assigned using the pair-wise comparison technique that is used in the Analytic Hierarchy Process (AHP). This technique forces the decision maker to choose between two objectives at a time by ranking them in terms of their relative contribution to the higher level objective.

The pair-wise comparisons were performed with the knowledge and understanding that Bruce A and Bruce B is to be operated for an additional 30 or more years, and that it must be operated safely and with high reliability to support electricity production per Bruce Power's business goals. The rankings used to distinguish between pairs of objectives are presented in Table 46.

Table 46: Ranking Definitions

Rank	Importance Descriptor
1	Equally
2	Equally – Moderately
3	Moderately
4	Moderately – Strongly
5	Strongly
6	Strongly – Very Strongly
7	Very Strongly
8	Very Strongly – Extremely
9	Extremely

In the following sections, the pairwise comparisons are presented in the form of tables, where

- the Tier 0 objective is supported by the Tier 1 Objectives (Section C.10.3);
- the Tier 1 objectives are supported by the Tier 2 Objectives (Section C.10.4);
- the Tier 2 objectives are supported by the Tier 3 Objectives (Section C.10.5).

For each comparison, a score is provided, based on the ranking definitions, followed by the justification for the score.

C.10.3. Tier 1 Pairwise Comparisons

C.10.3.1. Objective 1 vs. 2

1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	2	Enhanced confidence in the continued functional capability of Bruce A and Bruce B SSCs for an extended plant life
		7	

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards. A significant level of design improvement to meet modern standards will be required to enhance the design basis to a level comparable to those required for new NPPs. However, assessments of the design against modern versions of the original design requirements since the plants were put into operation have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth. In addition, safety upgrades and supplementary design and safety analyses are continually implemented to comply with the provisions of the PROL with high priority and as an integral part of continued safe and reliable operation.

In this comparison, the benefit of addressing enhanced confidence in the design basis of Bruce A and Bruce B SSCs is weighed against the importance of enhanced confidence in the continued functional capability of the SSCs for an extended plant life.

Improved knowledge about the actual condition of the SSCs and presence of permanent degradation through aging and taking the necessary preventative and corrective actions is essential to maintain the current design basis as prescribed in the PROL and to continue preventing initiating events that may lead to accidents or reduced plant reliability for long-term operation.

Given the limitations of the current plant configuration and the MCR schedule and the continuing improvements to the design basis as part of on-going compliance activities with the PROL, if resources have to be spent on either improving the design basis, or taking actions necessary to ensure confidence in the continued functional capability of the SSCs, the latter is favoured Very Strongly.

C.10.3.2. Objective 1 vs. 3

1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety
		5	

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards. A significant level of design improvement to meet modern standards will be required to enhance the design basis to a level comparable to those required for new NPPs. However, assessments of the design against modern versions of the original design requirements since the plants were put into operation have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth. In addition, safety upgrades and supplementary design and safety analyses are continually implemented to comply with the provisions of the PROL with high priority and as an integral part of continued safe and reliable operation.

In this comparison, the benefit of addressing enhanced confidence in the design basis of Bruce A and Bruce B SSCs is weighed against the importance of enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety.

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design improvement to meet modern codes and standards may be required to enhance confidence in the design basis for equipment qualification as compared to a new NPP. It is also noted that the assessments performed to date have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth. Since Bruce A and Bruce B was put into operation, hazard assessments, and associated design provisions for equipment qualification and scope of safety analysis has become a progressively more important area. This is due to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art hazard analysis methodologies based on new experimental data and OPEX.

Confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety will not only enhance confidence in the integrity of the safety analysis, but in general contribute very strongly to the Tier 0 objective of enhancing confidence in the safety of Bruce A and Bruce B SSCs under accident conditions or their robustness against internal and external hazards that were not considered in their original design. Such initiatives are also considered to be prerequisites to identifying which potential design improvements against modern codes and standards are more safety significant. Therefore, enhanced confidence in the equipment qualification of the Systems Important to Safety is favoured Strongly.

C.10.3.3. Objective 1 vs. 4

1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	4	Enhanced confidence in the safety analysis of Bruce A and Bruce B
		5	

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards. A significant level of design improvement to meet modern standards will be required to enhance the design basis to a level comparable to those required for new NPPs. However, assessments of the design against modern versions of the original design requirements since the plants were put into operation have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth. In addition, safety upgrades and supplementary design and safety analyses are continually implemented to comply with the provisions of the PROL with high priority and as an integral part of continued safe and reliable operation.

In this comparison, the benefit of addressing enhanced confidence in the design basis of Bruce A and Bruce B SSCs is weighed against the importance of enhanced confidence in the safety analysis of Bruce A and Bruce B.

A robust safety case relies on a comprehensive set of safety analyses that are based on modeling the actual configuration and condition of the plant, as well as use of state of the art tools and methodologies. Since Bruce A and Bruce B were put into operation, hazard assessments and the comprehensiveness of the scope of safety analysis has become a progressively more important area in confirming plant safety. This was mainly driven by the industry-wide responses to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art tools and analysis methodologies based on new experimental data and OPEX. Recently, update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area. It should also be noted that Bruce Power’s current nuclear safety assessment process and the safety analysis report meets the current regulatory requirements.

If resources have to be spent on either improving the design basis, or enhanced confidence in the safety analysis, the latter will make a stronger contribution to the Level 0 objective of enhancing confidence in the safety of Bruce A and Bruce B and reliability of electricity production. Taking actions necessary to ensure that the safety analyses are based on accurate information, validated state-of-the-art methods, and sound processes, is favoured Strongly. It is also noted that requirements to meet new CNSC regulatory documents and modern standards have been incorporated with respect to safety analysis in the PROL.

C.10.3.4. Objective 1 vs. 5

1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B
			2

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards. A significant level of design improvement to meet modern standards will be required to enhance the design basis to a level comparable to those required new NPPs. However, assessments of the design against modern versions of the original design requirements since the plants were put into operation have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth. In addition, safety upgrades and supplementary design and safety analyses are continually implemented to comply with the provisions of the PROL with high priority and as an integral part of continued safe and reliable operation.

In this comparison, the benefit of addressing enhanced confidence in the design basis of Bruce A and Bruce B SSCs is weighed against the importance of enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B.

Bruce Power has a robust and mature managed system that is built around the principle of ‘safety first’ as the overarching objective. The managed system has built-in continuous improvement features.

Given the acceptability for the design basis and the maturity of the managed system operation, it is judged that enhancements to the managed system will have a wider influence and potential benefits on many aspects of safe and reliable operation, whereas enhancing confidence in the design basis will have a benefit that is limited in its implementation. As such, spending resources on improving the design basis, or investing in action to improve the ability to operate and maintain an aging plant, the latter is favoured Equally – Moderately.

C.10.3.5. Objective 1 vs. 6

1	Enhanced confidence in the design basis of Bruce A and Bruce B SSCs	6	Enhanced confidence in the ability to maintain low environmental impact of Bruce A and Bruce B
		5	

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards. A significant level of design improvement to meet modern standards will be required to enhance the design basis to a level comparable to those required for new NPPs. However, assessments of the design against modern versions of the original design requirements since the plants were put into operation have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth. In addition, safety upgrades and supplementary design and safety analyses are continually implemented to comply with the provisions of the PROL with high priority and as an integral part of continued safe and reliable operation.

In this comparison, the benefit of addressing enhanced confidence in the design basis of the SSCs is weighed against the importance of enhanced confidence in the ability to maintain a low environmental impact of Bruce A and Bruce B.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of 'safety first' principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

Allocating resources in improving the capability to maintain a low environmental impact under normal operating conditions, as well as emergencies, is favoured Strongly. It is also noted that regulatory and public expectations with respect to maintaining a low environmental impact and improving environmental performance has been a prominent topic in recent years and during the licence hearings, more so than enhanced confidence in the design basis of the SSCs.

C.10.3.6. Objective 2 vs. 3

2	Enhanced confidence in the continued functional capability of Bruce A and Bruce B SSCs for an extended plant life	3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety
5			

Confidence in the continued functional capability of Bruce A and Bruce B SSCs to meet their current design and operating requirements through surveillance, testing, condition monitoring including follow-up maintenance repairs, replacements and modifications to the original design of SSCs in accordance with their design and operating envelope forms the basis of safe and reliable operation. Given the age of Bruce A and Bruce B and the considerations for extended plant life, eliminating gaps and enhancing confidence in the understanding of the current condition of Bruce A and Bruce B SSCs, taking necessary actions to ensure their continued functional capability and maintaining them in this state is considered to be the most important aspect of safe and reliable operation. In summary, enhanced confidence in the continued functional capability of the as-built plant is the pillar of event-free operation.

In this comparison, the benefit of addressing deficiencies in the knowledge about the actual condition of some of SSCs and taking actions to restore and enhance their functional capability is weighed against extent of the qualification of SSCs important to safety.

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design improvement to meet modern codes and standards may be required to enhance confidence in the design basis for equipment qualification as compared to a new NPP. It is also noted that the assessments performed to date have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth. Since Bruce A and Bruce B was put into operation, hazard assessments, and associated design provisions for equipment qualification and scope of safety analysis has become a progressively more important area. This is due to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art hazard analysis methodologies based on new experimental data and OPEX.

Improvement in knowledge about actual condition and ageing of equipment and follow-up maintenance repairs, replacements and modifications to the original design of SSCs in accordance with their design and operating envelope will reduce or ensure that the likelihood of events requiring operation of Systems Important to Safety have not increased since the plant was put into service. In this context the principle of defence-in-depth dictates that preventing events progressing into accidents will always be preferable to accident mitigation. Hence, resources allocated to preventing events with a higher probability due to poor functional capability is more beneficial than understanding the current capability of the design of SSCs important to safety due to initiating events with lower probability of occurrence. Therefore, enhanced confidence in continued functional capability is favoured Strongly.

C.10.3.7. Objective 2 vs. 4

2	Enhanced confidence in the continued functional capability of Bruce A and Bruce B SSCs for an extended plant life	4	Enhanced confidence in the safety analysis of Bruce A and Bruce B
5			

Confidence in the continued functional capability of Bruce A and Bruce B SSCs to meet their current design and operating requirements through surveillance, testing, condition monitoring including follow-up maintenance repairs, replacements and modifications to the original design of SSCs in accordance with their design and operating envelope forms the basis of safe and reliable operation. Given the age of Bruce A and Bruce B and the considerations for extended plant life, eliminating gaps and enhancing confidence in the understanding of the current condition of Bruce A and Bruce B SSCs, taking necessary actions to ensure their continued functional capability and maintaining them in this state is considered to be the most important aspect of safe and reliable operation. In summary, enhanced confidence in the continued functional capability of the as-built plant is the pillar of event-free operation.

In this comparison, the benefit of addressing deficiencies in knowledge about the actual condition of some of SSCs and taking actions to restore and enhance their functional capability is weighed against actions that will enhance confidence in the safety analysis.

A robust safety case relies on a comprehensive set of safety analyses that are based on modeling the actual configuration and condition of the plant, as well as use of state of the art tools and methodologies. Since Bruce A and Bruce B were put into operation, hazard assessments and the comprehensiveness of the scope of safety analysis has become a progressively more important area in confirming plant safety. This was mainly driven by industry-wide responses to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art tools and analysis methodologies based on new experimental data and OPEX. Recently, update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area. It should also be noted that Bruce Power's current nuclear safety assessment process and the safety analysis report meets the current regulatory requirements.

These two objectives are closely related, but conservative approaches used in safety analysis may to some extent compensate for uncertainties in the knowledge about the true condition of SSCs. In addition, it is a prerequisite to model the actual condition of the plant when performing safety analysis. It should also be noted that resources allocated to improving or restoring equipment functional capability provides more value as compared to demonstrate acceptable margins in cases of degraded functional capability. Therefore, enhanced confidence in continued functional capability is favoured Strongly.

C.10.3.8. Objective 2 vs. 5

2	Enhanced confidence in the continued functional capability of Bruce A and Bruce B SSCs for an extended plant life	5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B
8			

Confidence in the continued functional capability of Bruce A and Bruce B SSCs to meet their current design and operating requirements through surveillance, testing, condition monitoring including follow-up maintenance repairs, replacements and modifications to the original design of SSCs in accordance with their design and operating envelope forms the basis of safe and reliable operation. Given the age of Bruce A and Bruce B and the considerations for extended plant life, eliminating gaps and enhancing confidence in the understanding of the current condition of Bruce A and Bruce B SSCs, taking necessary actions to ensure their continued functional capability and maintaining them in this state is considered to be the most important aspect of safe and reliable operation. In summary, enhanced confidence in the continued functional capability of the as-built plant is the pillar of event-free operation.

In this comparison, the benefit of addressing deficiencies in knowledge about the actual condition of some of SSCs and taking actions to restore and enhance their functional capability is weighed against actions that will enhance confidence in the managed system for continued safe and reliable operation.

Bruce Power has a robust and mature managed system that is built around the principle of 'safety first' as the overarching objective. The managed system has built-in continuous improvement features.

Therefore, enhanced confidence in continued functional capability is favoured Very Strongly – Extremely.

C.10.3.9. Objective 2 vs. 6

2	Enhanced confidence in the continued functional capability of Bruce A and Bruce B SSCs for an extended plant life	6	Enhanced confidence in the ability to maintain low environmental impact of Bruce A and Bruce B
5			

Confidence in the continued functional capability of Bruce A and Bruce B SSCs to meet their current design and operating requirements through surveillance, testing, condition monitoring including follow-up maintenance repairs, replacements and modifications to the original design of SSCs in accordance with their design and operating envelope forms the basis of safe and reliable operation. Given the age of Bruce A and Bruce B and the considerations for extended plant life, eliminating gaps and enhancing confidence in the understanding of the current condition of Bruce A and Bruce B SSCs, taking necessary actions to ensure their continued functional capability and maintaining them in this state is considered to be the most important aspect of safe and reliable operation. In summary, enhanced confidence in the continued functional capability of the as-built plant is the pillar of event-free operation.

In this comparison, the benefit of addressing deficiencies in knowledge about the actual condition of some of SSCs and taking actions to restore and enhance their functional capability is weighed against the importance of actions to maintain radiological and non-radiological emissions as low as reasonably achievable during normal operations and to effectively respond to accidents and other emergencies that may have an adverse effect on the environment.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of ‘safety first’ principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

Addressing deficiencies in knowledge about the actual condition of some of SSCs and taking actions to restore and enhance their functional capability will reduce the likelihood of accidents and of unplanned releases during normal operations. Therefore, enhanced confidence in continued functional capability is favoured Strongly.

C.10.3.10. Objective 3 vs. 4

3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety	4	Enhanced confidence in the safety analysis of Bruce A and Bruce B
1			

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design improvement to meet modern codes and standards may be required to enhance confidence in the design basis for equipment qualification as compared to a new NPP. It is also noted that the assessments performed to date have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth.

A robust safety case relies on a comprehensive set of safety analyses that are based on modeling the actual configuration and condition of the plant, as well as use of state of the art tools and methodologies. Since Bruce A and Bruce B were put into operation, hazard assessments and the comprehensiveness of the scope of safety analysis has become a progressively more important area in confirming plant safety. This was mainly driven by industry-wide responses to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art tools and analysis methodologies based on new experimental data and OPEX. Recently, update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area. It should also be noted that Bruce Power's current nuclear safety assessment process and the safety analysis report meets the current regulatory requirements.

In this comparison the benefit of enhanced confidence in the equipment qualification of the Systems Important to Safety is weighed against the importance of enhanced confidence in the safety analysis.

These two objectives are complements of each other since the requirements for equipment qualification are based on the results of safety analyses that specify equipment qualification conditions as a result of imitating events and the safety analyses takes credit for the equipment qualification of Systems Important to Safety. Safety analysis therefore both relies on equipment qualification and sets requirements for it. Therefore, these objectives are favoured Equally.

C.10.3.11. Objective 3 vs. 5

3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety	5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B
5			

The original design basis of Bruce A and Bruce B does not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design improvement to meet modern codes and standards may be required to enhance confidence in the design basis for equipment qualification as compared to a new NPP. It is also noted that the assessments performed to date have shown that the original design of Bruce A and Bruce B, including the safety upgrades incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth. Since Bruce A was put into operation, hazard assessments, and associated design provisions for equipment qualification and scope of safety analysis has become a progressively more important area. This is due to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art hazard analysis methodologies based on new experimental data and OPEX.

In this comparison the benefit of enhanced confidence in the equipment qualification of the Systems Important to Safety is weighed against the importance of enhanced confidence in the managed system for continued safe and reliable operation.

Bruce Power has a robust and mature managed system that is built around the principle of ‘safety first’ as the overarching objective. The managed system has built-in continuous improvement features.

Therefore, given the maturity of Bruce Power’s managed system, enhanced confidence in equipment qualification of the Systems Important to Safety for continued safe operation is favoured Strongly.

C.10.3.12. Objective 3 vs. 6

3	Enhanced confidence in the equipment qualification of Bruce A and Bruce B Systems Important to Safety	6	Enhanced confidence in the ability to maintain a low environmental impact of Bruce A and Bruce B
1			

The original design basis of Bruce A does not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design improvement to meet modern codes and standards may be required to enhance confidence in the design basis for equipment qualification as compared to a new NPP. It is also noted that the assessments performed to date have shown that the original design of Bruce A, including the safety upgrades incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth.

Since Bruce A was put into operation, hazard assessments, and associated design provisions for equipment qualification and scope of safety analysis has become a progressively more important area. This is due to international OPEX and the resulting changes to design requirements for systems important to safety, as well as development of the state-of-the-art hazard analysis methodologies based on new experimental data and OPEX. Consequently, significant design improvements were implemented and assessments conducted to address design basis hazards in the recent years as part of the on-going compliance initiatives with the PROL. Some of these initiatives are still on-going and progress updates are provided to the CNSC on a regular basis.

In this comparison the benefit of enhanced confidence in the equipment qualification of the Systems Important to Safety is weighed against the importance of enhanced confidence in the ability to maintain a low environmental impact during normal operations and to effectively respond to accidents and other emergencies that may have an adverse effect on the environment.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of 'safety first' principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

Given Bruce Power's good environmental performance over the years, these objectives are favoured Equally.

C.10.3.13. Objective 4 vs. 5

4	Enhanced confidence in the safety analysis of Bruce A and Bruce B	5	Enhanced confidence in the managed system continued for safe and reliable operation of Bruce A and Bruce B
4			

A robust safety case relies on results of safety analyses based on a systematic and comprehensive set of postulated initiating events, state-of-the-art analysis methodologies based on up-to-date experimental data and OPEX and input data that reflects actual plant that is executed in accordance with procedures that meet applicable quality assurance requirements. Since Bruce A and Bruce B was put into operation, hazard assessments and the scope of safety analysis has become a progressively more important area as a result of international OPEX and the resulting changes to requirements for systems important to safety, as well as development of the state-of-the-art analysis methodologies based on new experimental data and OPEX. Recently, update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area. It should also be noted that Bruce Power's current nuclear safety assessment process and the safety analysis report meets the current regulatory requirements.

In this comparison the benefit of enhanced confidence in the safety analysis is weighed against the importance of enhanced confidence in the managed system continued for safe and reliable operation.

Bruce Power has a robust and mature managed system that is built around the principle of 'safety first' as the overarching objective. That managed system has built-in continuous improvement features.

Given the current state of Bruce Power's managed system and the need to upgrade safety analysis driven by plant ageing and new regulatory expectations, enhancing confidence in the safety analysis is favoured Moderately – Strongly.

C.10.3.14. Objective 4 vs. 6

4	Enhanced confidence in the safety analysis of Bruce A and Bruce B	6	Enhanced confidence in the ability to maintain a low environmental impact of Bruce A and Bruce B
1			

A robust safety case relies on results of safety analyses based on a systematic and comprehensive set of postulated initiating events, state-of-the-art analysis methodologies based on up-to-date experimental data and OPEX and input data that reflects actual plant that is executed in accordance with procedures that meet applicable quality assurance requirements. Since Bruce A and Bruce B was put into operation, hazard assessments and the scope of safety analysis has become a progressively more important area as a result of international OPEX and the resulting changes to requirements for systems important to safety, as well as development of the state-of-the-art analysis methodologies based on new experimental data and OPEX. Recently, update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area. It should also be noted that Bruce Power's current nuclear safety assessment process and the safety analysis report meets the current regulatory requirements.

In this comparison the benefit of enhanced confidence in the safety analysis is weighed against the importance of enhanced confidence in the ability to maintain a low environmental impact.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of 'safety first' principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

Given Bruce Power's good environmental performance over the years, these objectives are favoured Equally.

C.10.3.15. Objective 5 vs. 6

5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A and Bruce B	6	Enhanced confidence in the ability to maintain a low environmental impact of Bruce A and Bruce B
		4	

Bruce Power has a robust and mature managed system that is built around the principle of 'safety first' as the overarching objective. The managed system has built-in continuous improvement features. Bruce Power's managed system meets all the regulatory requirements in the PROL associated with management of the plant operations.

In this comparison the benefit of enhanced confidence in the managed system for continued safe and reliable operation is weighed against the ability to maintain a low environmental impact.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of 'safety first' principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

Therefore, given the maturity of Bruce Power's managed system, enhanced confidence in the ability to maintain a low environmental impact is favoured Moderately – Strongly

C.10.4. Tier 2 Pairwise Comparisons

C.10.4.1. Objective 1 Comparisons

C.10.4.1.1. Objective 1.1 vs. 1.2

1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1.2	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation
1			

Maintaining improved confidence in demonstration of conformance of the plant design basis with the applicable regulations, codes and standards together with the supporting documentation of the original design and changes made to date is the fundamental prerequisite in support of safe and reliable plant operation and the validity of safety analysis.

Since Bruce A and Bruce B was put into operation, upgrades to SSCs Important to Safety were implemented to meet CNSC expectations with respect to modern versions of applicable codes and standards in the PROL, as well as lessons learned from internal and external OPEX. Original plant configuration has also been modified as part of plant maintenance and repairs, as well as changes that have occurred due to ageing of SSCs. Therefore, it is equally important to ensure that actual configuration of the plant is within the design and operating envelope as described in the design documentation.

Hence, it is deemed that both activities feed back to each other as described under the current plant design governance in achieving enhanced confidence in the design basis. Therefore, these objectives are favoured Equally.

C.10.4.1.2. Objective 1.1 vs. 1.3

1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1.3	Enhanced confidence that the design of SSCs meets modern standards
6			

Maintaining improved confidence in demonstration of conformance of the plant design basis with the applicable regulations, codes and standards together with the supporting documentation of the original design and changes made to date is the fundamental prerequisite in support of safe and reliable plant operation and the validity of safety analysis.

Given that Bruce A and Bruce B has operated safely for many decades, it can be argued that the original design intent is sufficiently conservative and robust, even though it does not fully meet all the requirements of modern codes and standards. In addition, over the years upgrades to SSCs to Safety were implemented to meet CNSC expectations with respect to modern versions of applicable codes and standards, as well as lessons learned from internal and external OPEX.

Maintaining improved confidence in accurate knowledge and conformance of the design intent with the applicable regulations, codes and standards support both safe and reliable plant operation and the validity of safety analysis under aged conditions. Hence, it is deemed that it is more important to confirm the validity of the current design basis. Therefore, enhancing confidence that the actual design intent is accurately known is favoured Strongly – Very Strongly.

C.10.4.1.3. Objective 1.2 vs. 1.3

1.2	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	1.3	Enhanced confidence that the design of SSCs meets modern standards
6			

Since Bruce A and Bruce B was put into operation, upgrades to SSCs Important to Safety were implemented to meet CNSC expectations with respect to modern versions of applicable codes and standards in the PROL, as well as lessons learned from internal and external OPEX. Original plant configuration has also been modified as part of plant maintenance and repairs, as well as changes that have occurred due to ageing of SSCs. Therefore, maintaining enhanced confidence that the actual configuration of the plant is within the prescribed design and operating envelope as described in the design documentation is a fundamental prerequisite in support of safe and reliable plant operation and the validity of safety analysis on an on-going basis.

Given that Bruce A and Bruce B have operated safely for many decades, it can be argued that the original design intent is sufficiently conservative and robust, even though it does not fully meet all the requirements of modern codes and standards. In addition, over the years upgrades to SSCs to Safety were implemented to meet CNSC expectations with respect to modern versions of applicable codes and standards, as well as lessons learned from internal and external OPEX.

Maintaining improved confidence that the actual configuration of the SSCs meet the requirements described in the design documentation supports both safe and reliable plant operation and the validity of safety analysis under aged conditions. Hence, it is deemed that it is more important to confirm the actual configuration of the SSCs. Therefore, enhancing confidence that the actual configuration of the SSCs meet the requirements described in the design documentation is favoured Strongly – Very Strongly.

C.10.4.2. Objective 2 Comparisons

C.10.4.2.1. Objective 2.1 vs. 2.2

2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).	2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment Reliability Program.
		6	

These two objectives are complementary since knowledge about the actual condition of equipment is required before it can be determined whether functional capability can be restored. However, given the current state of Bruce A and Bruce B, and the programs in place to update the knowledge about the actual condition of equipment, it is considered to be a higher priority to restore the SSCs to a state that achieves the intended functionality. It should also be noted that Bruce Power has obtained a comprehensive set of data on the conditions of systems important to safety. Thus, it is more beneficial to take the necessary actions to restore SSCs to a state that achieves the intended functionality as compared to actions to improve knowledge about the condition of SSCs. In general, it can be categorically stated that it is more beneficial to correct known functionality issues first than to investigate for more.

Therefore, enhancing confidence in restoring SSCs to a state that achieves the intended functionality is favoured Strongly – Very Strongly.

C.10.4.2.2. Objective 2.1 vs. 2.3

2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).	2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life. This is achieved through the Maintenance Program activities.
		6	

These two objectives are also complementary since knowledge about actual condition of equipment is required before it can be determined what maintenance needs to be performed in the longer term. However, given the current state of Bruce A and Bruce B, and the programs in place to update the knowledge about the actual condition of equipment, it is considered to be a higher priority to maintain SSCs in a state that achieves reliable operation and safety performance. It should also be noted that Bruce Power has obtained a comprehensive set of data on the conditions of systems important to safety. Thus, it is more beneficial to take the necessary corrective actions to maintain SSCs in a state that achieves reliable operation and safety performance based on the information gathered as compared to actions to improve knowledge about the condition of SSCs. In general, it can be categorically stated that it is more beneficial to correct known maintenance issues first than to look for more.

Therefore, enhancing confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life is favoured Strongly – Very Strongly.

C.10.4.2.3. Objective 2.1 vs. 2.4

2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).	2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.
		4	

These two objectives are also complementary, since the knowledge about the actual condition of equipment is required before it can be determined whether operation is within the appropriate safe operating envelope. Although Bruce A and Bruce B is operated within a safe operating envelope as contained in the current operating documentation, completion of the Safe Operating Envelope (SOE) Project (in progress) will assure comprehensiveness and completeness of the operating documentation. However, given the current state of Bruce A and Bruce B programs in place to update the knowledge about the actual condition of equipment, it is considered to be a higher priority to enhance confidence of operation within the appropriate safe operating envelope. It should also be noted that Bruce Power has obtained a comprehensive set of data on the conditions of systems important to safety while SOE has not yet been implemented fully in all. Thus, it is more beneficial to take the necessary corrective actions to improve confidence of operation within the appropriate safe operating envelope as compared to actions to improve knowledge about the condition of SSCs. In general, it can be categorically stated that it is more beneficial to assure operation within a prescribed envelope rather than obtaining more information to establish one.

Therefore, enhancing confidence in operation within the appropriate safe operating envelope for an extended plant life is favoured Moderately – Strongly.

C.10.4.2.4. Objective 2.2 vs. 2.3

2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment Reliability Program.	2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life. This is achieved through the Maintenance Program activities.
			2

This comparison is in essence a choice between the need to maintain functional capability or performance and the need maintain structural integrity. Both are important, but the need to correct known failures and deterioration is more important than preventing potential functionality or performance issues. In general, maintaining functional capability does not assure structural integrity alone. Structural integrity is always a precondition for functionality and performance. However, maintaining functionality and performance indicates a level of structural integrity.

Therefore, enhancing confidence in maintaining SSCs in a state that achieves reliable operation and safety performance is favoured Equally – Moderately.

C.10.4.2.5. Objective 2.2 vs. 2.4

2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment Reliability Program.	2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.
	3		

These two objectives are complementary. The SOE scope is limited to intended functionality for systems important to safety and as such more focused on safety rather than reliable electricity production as an objective. It should also be noted that in order to support defence-in-depth, all equipment functionality issues that would lead to initiation of events that would require action of systems important to safety should be minimized. In this context, restoring SSCs to a state that achieves the intended functionality makes an important contribution to maintaining enhanced confidence in SOE. It is also noted that some SOE activities may drive activities in restoring SSCs to a state that achieves the intended functionality. Hence, SOE cannot be maintained without assuring functional capability first.

Therefore, enhancing confidence in the continued functional capability of the Bruce A and Bruce B SSCs is favoured Moderately.

C.10.4.2.6. Objective 2.3 vs. 2.4

2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life. This is achieved through the Maintenance Program activities.	2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.
4			

These two objectives are complementary. The SOE scope is limited to intended functionality for systems important to safety and as such more focused on safety rather than reliable electricity production as an objective. In this context, enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance has a larger scope and hence a greater impact. It should also be noted that in order to support defence-in-depth, all issues that impact safe and reliable operation of equipment that would lead to initiation of events which would require action of systems important to safety should be dealt with as an overriding priority. In this context, maintaining SSCs in a state that achieves reliable operation and safety performance also makes an important contribution to maintaining enhanced confidence in SOE. It is also noted that some SOE activities may drive activities in maintaining SSCs in a state that achieves reliable operation and safety performance. Hence, SOE cannot be maintained without assuring reliable operation and safety performance first.

Therefore, enhancing confidence in maintaining SSCs in a state that achieves reliable operation and safety performance is favoured Moderately – Strongly.

C.10.4.3. Objective 3 Comparisons

There is only one contribution to Objective 3, and therefore no pair-wise comparison is required.

C.10.4.4. Objective 4 Comparisons

C.10.4.4.1. Objective 4.1 vs. 4.2

4.1	Enhanced confidence in the comprehensiveness of the safety analysis	4.2	Enhanced confidence in conformance with the applicable safety analysis methods and associated acceptance criteria
4			

A robust safety case relies on results of safety analyses based on a systematic and comprehensive set of postulated initiating events, state-of-the-art analysis methodologies based on up-to-date experimental data and OPEX and input data that reflects actual plant that is executed in accordance with procedures that meet applicable quality assurance requirements.

Since Bruce A and Bruce B was put into operation, the scope of hazard assessments and safety analyses in terms of addressing Anticipated Operational Occurrences (AOOs) and Beyond Design Basis Accident (BDBAs) have expanded and became a progressively more important area as a result of international OPEX and the resulting changes to requirements for systems important to safety. Some progress has been made in developing plans to update Safety Reports to provide a more comprehensive scope based on a systematic treatment of initiating events.

Recently, an update of the safety analysis to address plant ageing and to meet new Canadian regulatory documents has led to a number of industry initiatives in development of state of the art methodologies to meet modern regulatory requirements and standards. In support of these initiatives, the Canadian Industry has also initiated a number of R&D projects in consolidation of the associated safety analysis acceptance criteria and supporting experiments. These initiatives improve the current technical basis of the criteria in place.

In this context, comprehensiveness of the current safety analysis is a more fundamental issue as compared to making improvements on the safety analysis methodologies, which to date have demonstrated inherent conservatisms and safety margins of the safety analyses contained in the Analysis of Record (AOR).

Therefore, enhancing confidence in the comprehensiveness of the safety analysis is favoured Moderately – Strongly.

C.10.4.5. Objective 5 Comparisons

C.10.4.5.1. Objective 5.1 vs. 5.2

5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience.	5.2	Enhanced confidence in the effectiveness of support for operators, maintainers and operations support staff
			5

Bruce Power has a mature staff selection and training process based on industry best practices, such as Systematic Approach to Training (SAT). Both processes and their implementation are continually improved based on internal audits, FASAs and industry OPEX.

Bruce A and Bruce B have a very comprehensive set of plant operating procedures and supporting documentation that has evolved and improved over the past 40+ years. Improvements in ensuring timely update of procedures and associated supporting documentation are being made, as this area has proven to be a challenge given the continuous improvements made to the plant. In addition, enhancements to improve human-machine interfaces are being implemented, as human factors issues based on OPEX and CNSC expectations have become a progressively important over the recent years.

Given the robustness of staff selection and training, enhancing confidence in the effectiveness of support for operators, maintainers and operations support staff is favoured Strongly.

C.10.4.5.2. Objective 5.1 vs. 5.3

5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience.	5.3	Enhanced confidence in a safe work environment
			4

Bruce Power has a mature staff selection and training process based on industry best practices, such as SAT. Both processes and their implementation are continually improved based on internal audits, FASAs and industry OPEX.

Bruce Power also has a strong conventional and radiological health and safety program, as well as a good performance record in both areas. Bruce Power Management continues its emphasis on improving the current performance in excellence in healthy and safe work environment as it is also very strongly related to maintaining a robust safety culture.

Given the robustness of staff selection and training, enhancing confidence in a safe work environment is favoured Moderately – Strongly.

C.10.4.5.3. Objective 5.1 vs. 5.4

5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience.	5.4	Enhanced confidence in organizational effectiveness
		3	

Bruce Power has a mature staff selection and training process based on industry best practices, such as SAT. Both processes and their implementation are continually improved based on internal audits, FASAs and industry OPEX.

Bruce Power also has a robust managed system that is built around the principle of 'safety first' as the overarching objective. The managed system has built-in continuous improvement features. As a result, governance and processes and their implementation have been steadily improved and matured over the years. However, implementation of the governance and processes is an area that requires constant oversight and improvement through the on-going performance monitoring and corrective action programs, which include those associated with enhancing staff capabilities.

Given the robustness of staff selection and training, enhancing confidence in organizational effectiveness is favoured Moderately.

C.10.4.5.4. Objective 5.2 vs. 5.3

5.2	Enhanced confidence in the effectiveness of support for operators, maintainers and operations support staff	5.3	Enhanced confidence in a safe work environment
1			

These two objectives are complementary. Bruce A and Bruce B have a very comprehensive set of plant operating procedures and supporting documentation that has evolved and improved over the past 40+ years. Improvements in ensuring timely update of procedures and associated supporting documentation are being made as this area has proven to be a challenge given the continuous improvements made to the plant. In addition, enhancements to improve human-machine interfaces are being implemented, as human factors issues based on OPEX and CNSC expectations have become a progressively important over the recent years.

Bruce Power also has a strong conventional and radiological health and safety program, as well as a good performance record in both areas. Bruce Power Management continues its emphasis on improving the current performance in excellence in a healthy and safe work environment, as it is also very strongly related to maintaining a robust safety culture.

Therefore, these objectives are favoured Equally.

C.10.4.5.5. Objective 5.2 vs. 5.4

5.2	Enhanced confidence in the effectiveness of support for operators, maintainers and operations support staff	5.4	Enhanced confidence in organizational effectiveness
3			

Bruce A and Bruce B have a very comprehensive set of plant operating procedures and supporting documentation that has evolved and improved over the past 40+ years. Improvements in ensuring timely update of procedures and associated supporting documentation are being made as this area has proven to be a challenge given the continuous improvements made to the plant. In addition, enhancements to improve human-machine interfaces are being implemented, as human factors issues based on OPEX and CNSC expectations have become a progressively important over the recent years.

Bruce Power also has a robust managed system that is built around the principle of ‘safety first’ as the overarching objective. The managed system has built-in continuous improvement features. As a result, governance and processes and their implementation have been steadily improved and matured over the years. However, implementation of the governance and processes is an area that requires constant oversight and improvement which includes enhancing confidence in the effectiveness of support for operators, maintainers and operations support staff.

Given the effectiveness of the managed system and the initiatives in place to improve timely updates of procedures and improvements in human factors, enhancing confidence in the effectiveness of support for operators, maintainers and operations support staff is favoured Moderately.

C.10.4.5.6. Objective 5.3 vs. 5.4

5.3	Enhanced confidence in a safe work environment	5.4	Enhanced confidence in organizational effectiveness
3			

Bruce Power has a strong conventional and radiological health and safety program, as well as a good performance record in both areas. Bruce Power Management continues its emphasis on improving the current performance in excellence in healthy and safe work environment as it is also very strongly related to maintaining a robust safety culture.

Bruce Power also has a robust managed system that is built around the principle of 'safety first' as the overarching objective. The managed system has built-in continuous improvement features. As a result, governance and processes and their implementation have been steadily improved and matured over the years. However, implementation of the governance and processes is an area that requires constant oversight and improvement which include enhancing safe work environment.

Given the effectiveness of the managed system and the initiatives in place to improve safe work environment staff, enhancing confidence in a safe work environment is favoured Moderately.

C.10.4.6. Objective 6 Comparisons

C.10.4.6.1. Objective 6.1 vs. 6.2

6.1	Enhanced confidence in maintaining a low environmental impact during normal operations	6.2	Enhanced confidence in the ability to mitigate releases associated with external/internal events or accidents
5			

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of the 'safety first' principle. In addition, regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. The recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

In this comparison, the importance of enhancing confidence in low environmental impact during normal operation is weighed against mitigating releases associated with postulated initiating events. In other words, the choice between events of lower frequency (accidental release due to initiating events) is weighed against potential releases that occur as part of normal plant operation, i.e., higher frequency.

Therefore, enhancing confidence in low environmental impact during normal operations is favoured Strongly.

C.10.5. Tier 3 Pairwise Comparisons

C.10.5.1. Objective 1.1 Comparisons

There is only one contribution to Objective 1.1, and therefore no pair-wise comparison is required.

C.10.5.2. Objective 1.2 Comparisons

There is only one contribution to Objective 1.2, and therefore no pair-wise comparison is required.

C.10.5.3. Objective 1.3 Comparisons

C.10.5.3.1. Objective 1.3.1 vs. 1.3.2

1.3.1	Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards	1.3.2	Enhanced confidence that the design analysis/qualification of the plant SSCs meet the enhanced or new analytical/qualification requirements included in the modern codes and standards
		7	

Enhanced or new analytical/qualification requirements included in the modern codes and standards enable a more accurate quantification of safety margins that were built into the current design and provide an improved understanding of the plant safety to establish if any new design features and provisions included in the modern codes and standards are warranted. As such, Objective 1.3.2 has a more primary role in assuring enhanced confidence that the design of SSCs meets modern standards.

Therefore, Objective 1.3.2 is favoured Very Strongly.

C.10.5.4. Objective 2.1 Comparisons

C.10.5.4.1. Objective 2.1.1 vs. 2.1.2

2.1.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) for an extended plant life	2.1.2	Enhanced confidence in knowledge about the current condition of SSCs Important to Reliability (SIR) for an extended plant life
7			

Although knowledge about the current condition of SSCs in the context of safety and reliability go hand in hand, any effort in understanding the condition of SSCs with respect to their safety function(s) is a much more important consideration. Moreover, SIS are a sub-set of SIR and hence enhanced confidence in the knowledge about their current condition contributes to enhanced confidence in knowledge about the current condition of SSCs Important to Reliability (SIR) for an extended plant life.

Therefore, Objective 2.1.1 is favoured Very Strongly, as it will also support Objective 2.1.2.

C.10.5.5. Objective 2.2 Comparisons

C.10.5.5.1. Objective 2.2.1 vs. 2.2.2

2.2.1	Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life	2.2.2	Enhanced confidence in restoring SIR to a state that achieves the intended functionality and extended plant life
7			

Although functional capability in the context of safety and reliability go hand in hand, any effort in restoring functional capability of SSCs and extended life with respect to their safety function(s) is a much more important consideration. It is also a fact that SIS are a sub-set of SIR and hence enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life equally contribute to Objective 2.2.2.

Therefore, Objective 2.2.1 is favoured Very Strongly, as it will also support Objective 2.2.2.

C.10.5.6. Objective 2.3 Comparisons

C.10.5.6.1. Objective 2.3.1 vs. 2.3.2

2.3.1	Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life	2.3.2	Enhanced confidence in maintaining SIR in a state that achieves reliable operation and safety performance and extended plant life
7			

Although maintenance activities in the context of safety and reliability go hand in hand, any effort in maintaining SSCs for reliability and extended life with respect to their safety function(s) is a much more important consideration. Moreover, SIS are a sub-set of SIR and hence enhanced confidence in maintaining SIS to a state that achieves reliable operation and safety performance for extended plant life equally contribute to Objective 2.3.2.

Therefore, Objective 2.3.1 is favoured Very Strongly, as it will also support Objective 2.3.2.

C.10.5.7. Objective 2.4 Comparisons

There is only one contribution to Objective 2.4, and therefore no pair-wise comparison is required.

C.10.5.8. Objective 3.1 Comparisons

C.10.5.8.1. Objective 3.1.1 vs. 3.1.2

3.1.1	Enhanced confidence in the current environmental qualification requirements of SIS resulting from deterministic safety analysis of Design Basis Accidents (DBAs)	3.1.2	Enhanced confidence in the equipment qualification requirements resulting from hazards analysis of internal and external events
		5	

Bruce Power has a mature EQ program based on the current deterministic safety analysis of DBAs. Maintenance of the EQ program has also been considered as part of equipment reliability and maintenance programs, including component ageing and obsolescence. Any improvements that may be required as a result of new hazards analysis of internal and external events that may be performed to meet current licensing requirements will enhance equipment qualification further.

Therefore, Objective 3.1.2 is favoured Strongly.

C.10.5.9. Objective 4.1 Comparisons

C.10.5.9.1. Objective 4.1.1 vs. 4.1.2

4.1.1	Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record	4.1.2	Enhanced confidence in the definition of initiating events and combinations thereof in the current analysis of record
4			

Objective 4.1.1 addresses the comprehensiveness of safety analysis pertaining to the contents of the AOR as a whole, in terms of the overall architecture of the AOR and elements that need to be included in each of the sections of each type of analysis. Objective 4.1.2 addresses the systematic approach to the definition of the safety analysis scope pertaining to the bases for the analyses that are included in the AOR, taking all initiating events and combinations thereof based on the modern requirements.

The current configuration and contents of the safety analysis represent an evolution and adaptation of requirements that were implemented on an as-needed basis over the years. There is a need to establish the extent to which the requirements of modern regulations, codes and standards as a whole are reflected in the current AOR, as this is a pre-requisite for meeting the Tier 2 Objective 4.1 (Enhanced confidence in the comprehensiveness of the safety analysis). Enhanced confidence in the definition of initiating events and combinations thereof in accordance with new requirements in the current AOR is a pre-requisite activity in meeting Objective 4.1. In this context, Objective 4.1.2 provides a more detailed account of enhanced confidence in meeting Objective 4.1, based on the new definition of initiating events and their combination thereof. Objective 4.1.1 provides an overall picture of the completeness of all elements in the current configuration of the AOR, which already includes some aspects of Objective 4.1.2.

Therefore, Objective 4.1.1 is favoured Moderately – Strongly.

C.10.5.9.2. Objective 4.1.1 vs. 4.1.3

4.1.1	Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record	4.1.3	Enhanced confidence in the coverage of all initiating events and combinations thereof of the current safety analysis of record
7			

Objective 4.1.1 addresses the comprehensiveness of safety analysis pertaining to the contents of the AOR as a whole, in terms of the overall architecture of the AOR and elements that need

to be included in each of the sections of each type of analysis. Objective 4.1.3 addresses the integrated coverage of all accidents in the Safety Report in accordance with new requirements, which pertains to the scope of the outstanding analyses that need to be included in the analysis of record taking all initiating events and combinations thereof based on the modern requirements.

The current configuration and contents of the safety analysis represent an evolution and adaptation of requirements that were implemented on an as-needed basis over the years. There is a need to establish the extent to which the requirements of modern regulations, codes and standards as a whole are reflected in the current AOR, as this is a pre-requisite for meeting the Tier 2 Objective 4.1 (Enhanced confidence in the comprehensiveness of the safety analysis). Enhanced confidence in the coverage of initiating events and combinations thereof in accordance with new requirements in the current AOR is the last step in the process of meeting Objective 4.1. In this context, Objective 4.1.3 provides a fully integrated account of the enhanced confidence in meeting Objective 4.1 based on new requirements. Objective 4.1.1 provides an overall picture of the completeness of all elements in the current configuration of the AOR, which already includes some aspects of Objective 4.1.3.

Therefore, Objective 4.1.1 is favoured Very Strongly.

C.10.5.9.3. Objective 4.1.2 vs. 4.1.3

4.1.2	Enhanced confidence in the definition of initiating events and combinations thereof in the current analysis of record	4.1.3	Enhanced confidence in the coverage of all initiating events and combinations thereof of the current safety analysis of record
	3		

Objective 4.1.2 addresses the systematic approach to the definition of the safety analysis scope pertaining to the bases for the analyses that are included in the AOR, taking all initiating events and combinations thereof based on the modern requirements. Objective 4.1.3 addresses the integrated coverage of all accidents in the Safety Report in accordance with new requirements, which pertains to the scope of the outstanding analyses that need to be included in the analysis of record taking all initiating events and combinations thereof based on the modern requirements.

Although equally important to the Tier 2 Objective 4.1, these Tier 3 sub-objectives must be sequenced in a manner that enables the effective implementation of improvements to achieve enhanced confidence. In this context, activities associated with Objective 4.1.2 must be completed first so that activities associated with Objective 4.1.3 can be implemented effectively.

Therefore, Objective 4.1.2 is favoured Moderately.

C.10.5.10. Objective 4.2 Comparisons

C.10.5.10.1. Objective 4.2.1 vs. 4.2.2

4.2.1	Enhanced confidence in the degree to which software used for accident analysis has been validated	4.2.2	Enhanced confidence in the degree to which acceptance criteria used in safety analysis is supported by experimental or operational data
		4	

The CANDU industry has invested significant resources on the validation of accident analysis software as associated QA requirements evolved over the years. As a result, safety analyses conducted in the recent years on issues where safety margins have been confirmed, utilized validated software. Software validation programs and initiatives are still being pursued based on feedback from the CNSC and industry priorities.

The technical bases for the acceptance criteria used in safety analysis have also been a priority. Research and development programs to establish limits for different sets of accidents have been initiated based on established gaps. A set of acceptance criteria based on the relevant phenomena and experimentally supported limits are essential elements for establishing confidence in safety margins and supporting software validation. In this context, Objective 4.2.2 is more fundamental than Objective 4.2.1 in supporting Tier 2 Objective 4.2.

Given that the industry has made more progress in Objective 4.2.1 than Objective 4.2.2, and that the results of Objective 4.2.2 can be applied immediately to the current analyses, Objective 4.2.2 is favoured Moderately – Strongly.

C.10.5.10.2. Objective 4.2.1 vs. 4.2.3

4.2.1	Enhanced confidence in the degree to which software used for accident analysis has been validated	4.2.3	Enhanced confidence in the application of modern methodologies and criteria in the conduct of safety analysis
4			

The CANDU industry has invested significant resources on the validation of accident analysis software as associated QA requirements evolved over the years. As a result, safety analyses conducted in the recent years on issues where safety margins have been confirmed, utilized validated software. Software validation programs and initiatives are still being pursued based on feedback from the CNSC and industry priorities.

Application of modern methodologies and criteria in the conduct of safety analysis has been progressively introduced by the CANDU industry as driven by OPEX and R&D findings. Industry-wide guidelines have been developed to standardize the use of these methodologies and secure regulatory acceptance. Improvements and changes to these methodologies are being made based on feedback from the CNSC, as well as their use in licensing applications. In this context, Objective 4.2.1 is more fundamental than Objective 4.2.3, and the results of Objective 4.2.1 are in more immediate need of improvement for conducting future analyses.

Given that the industry has made more progress in Objective 4.2.3 than Objective 4.2.1, and that the results of Objective 4.2.1 can be applied immediately in planned analyses, Objective 4.2.1 is favoured Moderately – Strongly.

C.10.5.10.3. Objective 4.2.2 vs. 4.2.3

4.2.2	Enhanced confidence in the degree to which acceptance criteria used in safety analysis is supported by experimental or operational data	4.2.3	Enhanced confidence in the application of modern methodologies and criteria in the conduct of safety analysis
8			

Technical bases for the acceptance criteria used in safety analysis have been a CANDU Industry priority. Research and development programs to establish limits for different sets of accidents have been initiated based on established gaps. A set of acceptance criteria based on the relevant phenomena and experimentally supported limits are essential elements for establishing confidence in safety margins and supporting software validation.

Application of modern methodologies and criteria in the conduct of safety analysis has been progressively introduced by the CANDU industry as driven by OPEX and R&D findings. Industry-wide guidelines have been developed to standardize the use of these methodologies and secure regulatory acceptance. Improvements and changes to these methodologies are being made based on feedback from the CNSC, as well as their use in licensing applications. In this context, Objective 4.2.2 is more fundamental than Objective 4.2.3, and the results of Objective 4.2.2 are in more immediate need of improvement for conducting future analyses.

Given that the industry has made more progress in Objective 4.2.3 than Objective 4.2.2, and that the results of Objective 4.2.2 can be applied immediately in planned analyses, Objective 4.2.2 is favoured Very Strongly – Extremely.

C.10.5.11. Objective 5.1 Comparisons

C.10.5.11.1. Objective 5.1.1 vs. 5.1.2

5.1.1	Enhanced confidence in the selection and training of staff	5.1.2	Enhanced confidence in the dissemination and assimilation of internal and external operating experience
3			

Bruce Power has a mature staff selection and training program, as well as an OPEX program. Both programs are being continuously improved and receive significant management focus. However, selection and training of staff is a more fundamental requirement than dissemination and assimilation of internal and external OPEX, which is also an integral part of training of staff.

Therefore, Objective 5.1.1 is favoured Moderately.

C.10.5.12. Objective 5.2 Comparisons

C.10.5.12.1. Objective 5.2.1 vs. 5.2.2

5.2.1	Enhanced confidence in the comprehensiveness and effectiveness of procedures	5.2.2	Enhanced confidence in the appropriateness, validity and timeliness of plant and process information
			5

Bruce Power has comprehensive and effective governance supported by detailed procedures covering all aspects of plant operation. These procedures are updated on a regular basis based on feedback on their adequacy and use, as well as internal/external OPEX. FASA and audits of functional groups also evaluate procedural use and adherence and provide feedback on weaknesses and areas for improvement.

Bruce A and Bruce B have a very comprehensive set of plant and process information that has been compiled and improved over the past 40+ years. Improvements in ensuring the timely update of plant and process information are being made, as this area has proven to be a challenge given the continuous improvements made to the plant.

Given the maturity of Objective 5.2.1 compared to Objective 5.2.2 and the on-going need to keep up with updating changes to the plant, Objective 5.2.2 is favoured Strongly.

C.10.5.12.2. Objective 5.2.1 vs. 5.2.3

5.2.1	Enhanced confidence in the comprehensiveness and effectiveness of procedures	5.2.3	Enhanced confidence in the appropriateness of plant control interfaces (human factors)
			7

Bruce Power has comprehensive and effective governance supported by detailed procedures covering all aspects of plant operation. These procedures are updated on a regular basis based on feedback on their adequacy and use, as well as internal/external OPEX. FASA and audits of functional groups also evaluate procedural use and adherence and provide feedback on weaknesses and areas for improvement.

Bruce A and Bruce B have a well established set of plant control interfaces developed over the years. In recent years, enhancements to improve human-machine interfaces are being pursued and implemented, as human factors issues based on OPEX and CNSC expectations have become progressively more important.

Given the maturity of Objective 5.2.1 compared to Objective 5.2.3 and the on-going need for improvements in improving plant control interfaces, Objective 5.2.3 is favoured Very Strongly.

C.10.5.12.3. Objective 5.2.2 vs. 5.2.3

5.2.2	Enhanced confidence in the appropriateness, validity and timeliness of plant and process information	5.2.3	Enhanced confidence in the appropriateness of plant control interfaces (human factors)
		3	

Bruce Power has comprehensive and effective governance supported by detailed procedures covering all aspects of plant operation. These procedures are updated on a regular basis based on feedback on their adequacy and use, as well as internal/external OPEX. FASA and audits of functional groups also evaluate procedural use and adherence and provide feedback on weaknesses and areas for improvement.

Bruce A and Bruce B have a very comprehensive set of plant and process information that has been compiled and improved over the past 40+ years. Improvements in ensuring timely update of plant and process information are being made as this area has proven to be a challenge given the continuous improvements made to the plant.

Bruce A and Bruce B have a well established set of plant control interfaces developed over the years. In recent years, enhancements to improve human-machine interfaces are being pursued and implemented, as human factors issues based on OPEX and CNSC expectations have become progressively more important.

Given the maturity of Objective 5.2.2 compared to Objective 5.2.3 and the on-going need for improvements in improving plant control interfaces, Objective 5.2.3 is favoured Moderately.

C.10.5.13. Objective 5.3 Comparisons

C.10.5.13.1. Objective 5.3.1 vs. 5.3.2

5.3.1	Enhanced confidence in radiation protection	5.3.2	Enhanced confidence in conventional health and safety
3			

Bruce Power has a very strong conventional health and safety program, as well as an excellent performance record. Bruce Power's radiological health and safety program is also strong and there are numerous initiatives to improve it as units have undergone long-term operational changes, such as lay-ups and refurbishment of major components, which require additional improvements to deal with associated radiological hazards.

Bruce Power Management continues its emphasis on improving the current performance of excellence in healthy and safe work environment in both areas, as it is also very strongly related to maintaining a robust safety culture.

Given the maturity of Objective 5.3.2 compared to Objective 5.3.1 and the on-going need for improvements in enhanced confidence in radiation protection, Objective 5.3.1 is favoured Moderately.

C.10.5.14. Objective 5.4 Comparisons

C.10.5.14.1. Objective 5.4.1 vs. 5.4.2

5.4.1	Enhanced confidence in management system structure and processes	5.4.2	Enhanced confidence in safety culture
			4

Bruce Power has a very robust management system and processes that has matured since its inception. Safety culture has been one of the cornerstones of the overall governance, and is a living attribute of the organization that must be improved on a continuous basis.

Therefore, Objective 5.4.2 is favoured Moderately – Strongly.

C.10.5.14.2. Objective 5.4.1 vs. 5.4.3

5.4.1	Enhanced confidence in management system structure and processes	5.4.3	Enhanced confidence in performance monitoring and corrective action
			7

Bruce Power has a very robust management system and processes that has matured since its inception. Performance monitoring activities have been well established and executed, although follow-up actions need improvement based on the assessment of follow-up reviews of audits and FASAs.

Therefore, Objective 5.4.3 is favoured Very Strongly.

C.10.5.14.3. Objective 5.4.2 vs. 5.4.3

5.4.2	Enhanced confidence in safety culture	5.4.3	Enhanced confidence in performance monitoring and corrective action
		3	

Safety culture has been one of the cornerstones of the overall governance, and is a living attribute of the organization that must be improved on a continuous basis. Performance monitoring activities have been well established and executed, although follow-up actions need improvement based on the assessment of follow-up reviews of audits and FASAs. Excellence in completion of follow-ups to performance monitoring and corrective action is also a very strong contributor and indicator of safety culture.

Therefore, Objective 5.4.3 is favoured Moderately.

C.10.5.15. Objective 6.1 Comparisons

C.10.5.15.1. Objective 6.1.1 vs. 6.1.2

6.1.1	Enhanced confidence in low impact of radioactive releases	6.1.2	Enhanced confidence in low impact of non-radiological releases
7			

Consequences of radiological releases and their long-term effects are more severe as compared to the potential non-radiological releases associated with Bruce A and Bruce B operations.

Therefore, Objective 6.1.1 is favoured Very Strongly.

C.10.5.16. Objective 6.2 Comparisons

C.10.5.16.1. Objective 6.2.1 vs. 6.2.2

6.2.1	Enhanced confidence in the ability to mitigate releases associated with anticipated operational occurrences and design basis events	6.2.2	Enhanced confidence in the ability to mitigate releases associated with beyond design basis events
5			

The likelihood of a DBA occurring is at least an order of magnitude higher than a BDBA, although the potential consequences of BDBAs can be more severe. It is also noted that enhanced confidence Objective 6.2.1 also supports enhanced confidence in Objective 6.2.2.

Therefore, Objective 6.2.1 is favoured Strongly.

Appendix D – CATEGORY 1: No Reasonable and Practicable Improvements can be Identified

Appendix D consists of those micro-gaps identified in the Safety Factor Reports for which no reasonable or practicable improvements can be identified.

- Table 47 provides a consolidation of all micro-gaps within this category. It is ordered such that gaps that are similar or identical appear consecutively. This can be regarded as a "smart table of contents" for the micro-gaps discussed in the next bullet, and provides a direct linkage back to the origin of the micro-gaps in the Safety Factor Reports.
- Table 48 provides the details for each of the micro-gaps within this category. This is based on an export from the PSR database, and is ordered first by Safety Factor, then by regulatory document/code/standard, then by clause.

The micro-gap number, which is provided in both tables, facilitates their use.

Table 47: Consolidation of Micro-gaps with No Reasonable and Practicable Improvements

Category 1- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_CNSC REGDOC 2.5.2_4.2.2_15	SF01-02-15	CNSC REGDOC 2.5.2	4.2.2	6
SF01_CNSC REGDOC 2.5.2_4.2.2_16	SF01-02-16	CNSC REGDOC 2.5.2	4.2.2	6
SF05_CNSC REGDOC 2.5.2_4.2.2_15	SF05-06-15	CNSC REGDOC 2.5.2	4.2.2	6
SF05_CNSC REGDOC 2.5.2_4.2.2_16	SF05-10-16	CNSC REGDOC 2.5.2	4.2.2	6
SF06_CNSC REGDOC 2.5.2_4.2.2_15	SF06-01-15	CNSC REGDOC 2.5.2	4.2.2	6
SF06_CNSC REGDOC 2.5.2_4.2.2_16	SF06-01-16	CNSC REGDOC 2.5.2	4.2.2	6
SF01_CNSC REGDOC 2.5.2_6.1.1_16	SF01-03-16	CNSC REGDOC 2.5.2	6.1.1	11
SF01_CNSC REGDOC 2.5.2_6.1.1_15	SF01-03-15	CNSC REGDOC 2.5.2	6.1.1	11
SF01_CSA N290.11-13_5.2.2.10_15	SF01-03-15	CSA N290.11-13	5.2.2.10	11
SF01_CSA N290.11-13_5.2.2.10_16	SF01-03-16	CSA N290.11-13	5.2.2.10	11
SF01_CNSC REGDOC 2.5.2_6.4_16	SF01-01-16	CNSC REGDOC 2.5.2	6.4	12
SF01_CNSC REGDOC 2.5.2_6.4_15	SF01-01-15	CNSC REGDOC 2.5.2	6.4	12
SF05_CNSC REGDOC 2.5.2_4.2.3_16	SF05-02-16	CNSC REGDOC 2.5.2	4.2.3	12
SF05_CNSC REGDOC 2.5.2_6.4_16	SF05-02-16	CNSC REGDOC 2.5.2	6.4	12
SF01_CNSC REGDOC 2.5.2_7.3.2_15	SF01-01-15	CNSC REGDOC 2.5.2	7.3.2	15
SF01_CNSC REGDOC 2.5.2_7.3.2_16	SF01-01-16	CNSC REGDOC 2.5.2	7.3.2	15
SF01_CNSC REGDOC 2.5.2_7.4.1_16	SF01-01-16	CNSC REGDOC 2.5.2	7.4.1	16
SF01_CNSC REGDOC 2.5.2_7.4.1_15	SF01-01-15	CNSC REGDOC 2.5.2	7.4.1	16
SF01_CNSC REGDOC 2.5.2_7.5_15	SF01-01-15	CNSC REGDOC 2.5.2	7.5	17

Category 1- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_CNSC REGDOC 2.5.2_7.5_16	SF01-01-16	CNSC REGDOC 2.5.2	7.5	17
SF01_CNSC REGDOC 2.5.2_7.6.2_15	SF01-05-15	CNSC REGDOC 2.5.2	7.6.2	18
SF01_CNSC REGDOC 2.5.2_7.6.2_16	SF01-05-16	CNSC REGDOC 2.5.2	7.6.2	18
SF01_CSA N290.0-11_4.5-4.8_16	SF01-05-16	CSA N290.0-11	4.5-4.8	18
SF01_CSA N290.0-11_4.5-4.8_15	SF01-05-15	CSA N290.0-11	4.5-4.8	18
SF01_CNSC REGDOC 2.5.2_7.6.2_16	SF01-03-16	CNSC REGDOC 2.5.2	7.6.2	19
SF01_CNSC REGDOC 2.5.2_7.6.5.2_16	SF01-20-16	CNSC REGDOC 2.5.2	7.6.5.2	21
SF01_CNSC REGDOC 2.5.2_7.10_16	SF01-09-16	CNSC REGDOC 2.5.2	7.10	23
SF01_CNSC REGDOC 2.5.2_7.13.1_16	SF01-01-16	CNSC REGDOC 2.5.2	7.13.1	26
SF01_CNSC REGDOC 2.5.2_7.13.1_15	SF01-01-15	CNSC REGDOC 2.5.2	7.13.1	26
SF01_CNSC REGDOC 2.5.2_7.13.1_16	SF01-16-16	CNSC REGDOC 2.5.2	7.13.1	27
SF07_CNSC REGDOC 2.5.2_7.13.1_16	SF07-01-16	CNSC REGDOC 2.5.2	7.13.1	27
SF01_CNSC REGDOC 2.5.2_7.15.1_16	SF01-01-16	CNSC REGDOC 2.5.2	7.15.1	28
SF01_CNSC REGDOC 2.5.2_7.15.1_15	SF01-03-15	CNSC REGDOC 2.5.2	7.15.1	28
SF01_CNSC REGDOC 2.5.2_7.15.3_16	SF01-10-16	CNSC REGDOC 2.5.2	7.15.3	29
SF01_CNSC REGDOC 2.5.2_7.15.3_15	SF01-10-15	CNSC REGDOC 2.5.2	7.15.3	29
SF01_CNSC REGDOC 2.5.2_8.1_16	SF01-01-16	CNSC REGDOC 2.5.2	8.1	32
SF01_CNSC REGDOC 2.5.2_8.1_15	SF01-01-15	CNSC REGDOC 2.5.2	8.1	32
SF01_CNSC REGDOC 2.5.2_8.3.2_15	SF01-01-15	CNSC REGDOC 2.5.2	8.3.2	34
SF01_CNSC REGDOC 2.5.2_8.3.2_16	SF01-01-16	CNSC REGDOC 2.5.2	8.3.2	34
SF01_CNSC REGDOC 2.5.2_8.4.2_15	SF01-05-15	CNSC REGDOC 2.5.2	8.4.2	36
SF06_CNSC REGDOC 2.5.2_8.4.2_15	SF06-02-15	CNSC REGDOC 2.5.2	8.4.2	36
SF01_CNSC REGDOC 2.5.2_8.6.12_15	SF01-11-15	CNSC REGDOC 2.5.2	8.6.12	37
SF01_CNSC REGDOC 2.5.2_8.6.12_16	SF01-11-16	CNSC REGDOC 2.5.2	8.6.12	37
SF01_CNSC REGDOC 2.5.2_8.6.12_15	SF01-01-15	CNSC REGDOC 2.5.2	8.6.12	38
SF01_CNSC REGDOC 2.5.2_8.8_15	SF01-11-15	CNSC REGDOC 2.5.2	8.8	39
SF01_CNSC REGDOC 2.5.2_8.8_16	SF01-23-16	CNSC REGDOC 2.5.2	8.8	39
SF01_CNSC REGDOC 2.5.2_8.8_16	SF01-23-16	CNSC REGDOC 2.5.2	8.8	39
SF01_CNSC REGDOC 2.5.2_8.10.3_16	SF01-22-16	CNSC REGDOC 2.5.2	8.10.3	43
SF01_CNSC REGDOC 2.5.2_8.10.4_16	SF01-09-16	CNSC REGDOC 2.5.2	8.10.4	44
SF01_CNSC REGDOC 2.5.2_8.10.4_15	SF01-07-15	CNSC REGDOC 2.5.2	8.10.4	44
SF05_CSA REGDOC 2.5.2_8.10.4_15	SF05-02-15	CNSC REGDOC 2.5.2	8.10.4	44
SF05_CNSC REGDOC 2.5.2_8.10.4_16	SF05-08-16	CNSC REGDOC 2.5.2	8.10.4	44
SF01_CNSC REGDOC 2.5.2_8.12.2_16	SF01-13-16	CNSC REGDOC 2.5.2	8.12.2	45
SF01_CNSC REGDOC 2.5.2_8.12.2_15	SF01-13-15	CNSC REGDOC 2.5.2	8.12.2	45
SF01_CNSC REGDOC 2.5.2_10.1_16	SF01-14-16	CNSC REGDOC 2.5.2	10.1	54
SF01_CNSC REGDOC 2.5.2_10.1_15	SF01-14-15	CNSC REGDOC 2.5.2	10.1	54
SF01_CSA N290.0-11_4.1-4.4_15	SF01-01-15	CSA N290.0-11	4.1-4.4	56

Category 1- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_CSA N290.0-11_4.1-4.4_16	SF01-01-16	CSA N290.0-11	4.1-4.4	56
SF01_CSA N290.0-11_4.5-4.8_16	SF01-20-16	CSA N290.0-11	4.5-4.8	57
SF01_CSA N290.0-11_4.5-4.8_15	SF01-01-15	CSA N290.0-11	4.5-4.8	57
SF01_CSA N290.1_4.2.6_16	SF01-20-16	CSA N290.1	4.2.6	57
SF01_CSA N290.0-11_4.9-4.13_16	SF01-01-16	CSA N290.0-11	4.9-4.13	60
SF01_CSA N290.0-11_4.9-4.13_16	SF01-01-16	CSA N290.0-11	4.9-4.13	60
SF01_CSA N290.0-11_4.9-4.13_15	SF01-01-15	CSA N290.0-11	4.9-4.13	60
SF01_CSA N290.3-11_5.5_5.7_15	SF01-11-15	CSA N290.3-11	5.5_5.7	65
SF01_CSA N290.3-11_A.2.3_16	SF01-19-16	CSA N290.3-11	A.2.3	66
SF01_CSA N290.3-11_10.1_15	SF01-03-15	CSA N290.3-11	10.1	67
SF01_CSA N290.3-11_10.1_16	SF01-03-16	CSA N290.3-11	10.1	67
SF01_CSA N290.3-11_A.2.5_16	SF01-19-16	CSA N290.3-11	A.2.5	68
SF01_CSA N290.3-11_14.1_15	SF01-05-15	CSA N290.3-11	14.1	69
SF01_CSA N290.3-11_A.3.1_16	SF01-19-16	CSA N290.3-11	A.3.1	70
SF01_CSA N290.3-11_A.3.4_16	SF01-19-16	CSA N290.3-11	A.3.4	71
SF05_CNSC REGDOC 2.5.2_6.4_15	SF05-02-15	CNSC REGDOC 2.5.2	6.4	129
SF05_CNSC REGDOC 2.5.2_6.4_16	SF05-02-16	CNSC REGDOC 2.5.2	6.4	129

Table 48: Micro-gaps with No Reasonable and Practicable Improvements

Gap #	SF01_CNSC REGDOC 2.5.2_10.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	10.1 Design for environmental protection
Requirement Assessed	The design shall make adequate provision to protect the environment and to mitigate the impact of the NPP on the environment. A review of the design shall confirm that this provision has been met. A systematic approach shall be used to assess the potential biophysical environmental effects of the NPP on the environment, and the effects of the environment on the NPP. Guidance The design should incorporate the “best available technology and techniques economically achievable” (BATEA) principle for aspects of the design related to environmental protection.
Macro-Gap	SF01-14-15
Issue/Gap Description	Section 2 Site Description of Part 1 of the Safety Report [NK21-SR-01320-00001, Rev. 005] describes the potential effect of the plant on population, agriculture, industry, transportation, fishing and recreation. The Bruce A design does not incorporate the best available technology and techniques economically achievable principle as recommended in the guidance section. Therefore, it is assessed as a gap (Gap).
Rationale	Bruce A is in compliance with the current licensing basis which makes adequate provisions for the protection of the environment through various means such as pollution prevention, ALARA and BATEA. Such a requirement is considered to be specifically applicable to a new nuclear plant. Application of BATEA principle is a fundamental design criteria which impacts the whole plant and hence impracticable to implement. Any future design changes or modifications to the plant which impact the environment will take BATEA into account. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_10.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	10.1 Design for environmental protection
Requirement Assessed	The design shall make adequate provision to protect the environment and to mitigate the impact of the NPP on the environment. A review of the design shall confirm that this provision has been met. A systematic approach shall be used to assess the potential biophysical environmental effects of the NPP on the environment, and the effects of the environment on the NPP. Guidance The design should incorporate the “best available technology and techniques economically achievable” (BATEA) principle for aspects of the design related to environmental protection.
Macro-Gap	SF01-14-16
Issue/Gap Description	Section 2 Site Description of Part 1 of the Safety Report [NK29-SR-01320-00001, R005] describes the potential effect of the plant on population, agriculture, industry, transportation, fishing and recreation. The Bruce B design does not incorporate the best available technology and techniques economically achievable principle as recommended in the guidance section. Therefore, it is assessed as a gap (Gap).
Rationale	Bruce B is in compliance with the current licensing basis which makes adequate provisions for the protection of the environment through various means such as pollution prevention, ALARA and BATEA. Such a requirement is considered to be specifically applicable to a new nuclear plant. Application of BATEA principle is a fundamental design criteria which impacts the whole plant and hence impracticable to implement. Any future design changes or modifications to the plant which impact the environment will take BATEA into account. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.2 Safety goals
Requirement Assessed	Qualitative safety goals A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. Quantitative application of the safety goals For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency 2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. Core damage frequency The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-5 per reactor year. Small release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E15 becquerels of iodine-131 shall be less than 1E-5 per reactor year. A greater release may require temporary

	evacuation of the local population. Large release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E14 becquerels of cesium-137 shall be less than 1E-6 per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.
Macro-Gap	SF01-02-15
Issue/Gap Description	A review of the same clause in RD-337 indicated that the Bruce A safety goals are less restrictive (i.e., larger) than those proposed for new plants [NK21-CORR-00531-11005]. This is identified as a gap (Gap).
Rationale	Bruce A is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement. The more restrictive requirements are considered to be specifically

	applicable to a new nuclear plant as such goals dictate the technical basis of NPP design starting from siting to decommissioning. Originally, Bruce A and Bruce B SSCs were not designed to meet the more stringent Safety Goal Limits as prescribed for new NPP design. Such limits affect the plant design as a whole and would have required a fundamentally different approach in the design of SSCs and various options to be pursued to meet the more stringent safety goals. It is judged that it is impracticable to implement such a change at the plant level. However, Bruce Power continues to make practicable design changes to improve plant safety and safety analysis margins including the current safety goals. For example, as demonstrated in SFR-6, PSA results based on the improvements made as a result of follow-up actions and initiatives to the Fukushima event for both Bruce A and Bruce B. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.2 Safety goals
Requirement Assessed	Qualitative safety goals A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. Quantitative application of the safety goals For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency 2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. Core damage frequency The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-5 per reactor year. Small release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E15 becquerels of iodine-131 shall be less than 1E-5 per reactor year. A greater release may require temporary

	evacuation of the local population. Large release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E14 becquerels of cesium-137 shall be less than 1E-6 per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.
Macro-Gap	SF01-02-16
Issue/Gap Description	Although the result of each individual PRA meets the safety goal limits set up for Bruce B PRAs (with the exception of high wind LRF result as noted above), their aggregates obtained by respective summation of SCDFs and LRFs do not meet the more stringent quantitative safety goal targets set up in the requirement clause. Therefore, a gap is assessed against this clause (gap).
Rationale	Bruce B is in compliance with the current licensing basis which aligns with

	the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement. The more restrictive requirements are considered to be specifically applicable to a new nuclear plant as such goals dictate the technical basis of NPP design starting from siting to decommissioning. Originally, Bruce A and Bruce B SSCs were not designed to meet the more stringent Safety Goal Limits as prescribed for new NPP design. Such limits affect the plant design as a whole and would have required a fundamentally different approach in the design of SSCs and various options to be pursued to meet the more stringent safety goals. It is judged that it is impracticable to implement such a change at the plant level. However, Bruce Power continues to make practicable design changes to improve plant safety and safety analysis margins including the current safety goals. For example, as demonstrated in SFR-6, PSA results based on the improvements made as a result of follow-up actions and initiatives to the Fukushima event for both Bruce A and Bruce B. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_6.1.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.1.1 Physical barriers
Requirement Assessed	To ensure the overall safety concept of defence in depth is maintained, the design shall provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment. Such barriers shall include the fuel matrix, the fuel cladding, the reactor coolant pressure boundary, and the containment. In addition, the design shall provide for an exclusion zone. To the extent practicable, the design shall prevent: 1. challenges to the integrity of physical barriers 2. failure of a barrier when challenged 3. failure of a barrier as a consequence of failure of another barrier 4. the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences The design shall also allow for the fact that the existence of multiple levels of defence does not normally represent a sufficient basis for continued power operation in the absence of one defence level.
Macro-Gap	SF01-03-15
Issue/Gap Description	The list of internal initiating events is presented in Table 2-1 of Part 3 of the Safety Report; however events initiated as a result of human errors in operation and maintenance are not explicitly identified. Initiating event frequencies include implicitly any relevant operator error that may cause the initiating event. Therefore, this is identified as a gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce A and B Safety Reports. Bruce A and Bruce B designs incorporate engineered barriers and features to prevent failures from errors in operation and maintenance that could result in harmful consequences in terms of redundancy, diversity and separation. However, these provisions were not always explicitly defined as requirements in terms of errors in operation and maintenance. As a general requirement it is not possible to make additional wholesale design changes to prevent the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences. In addition, Operating Policies and Principles and supporting operating documentation and operating and maintenance procedures provide additional barriers to minimize the likelihood of events initiated as a result of operator errors or errors in maintenance.

	Capability of the current design of SSCs resulting from human errors will be analyzed as part of AI 090739 under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. Errors in maintenance are also considered as leading to equipment failures which are covered under PIEs and the event classification. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_6.1.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.1.1 Physical barriers
Requirement Assessed	To ensure the overall safety concept of defence in depth is maintained, the design shall provide multiple physical barriers to the uncontrolled release of radioactive materials to the environment. Such barriers shall include the fuel matrix, the fuel cladding, the reactor coolant pressure boundary, and the containment. In addition, the design shall provide for an exclusion zone. To the extent practicable, the design shall prevent: 1. challenges to the integrity of physical barriers 2. failure of a barrier when challenged 3. failure of a barrier as a consequence of failure of another barrier 4. the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences The design shall also allow for the fact that the existence of multiple levels of defence does not normally represent a sufficient basis for continued power operation in the absence of one defence level.
Macro-Gap	SF01-03-16
Issue/Gap Description	The list of internal initiating events is presented in Table 2-1 of Part 3 of the Safety Report; however events initiated as a result of human errors in operation and maintenance are not explicitly identified. Initiating event frequencies include implicitly any relevant operator error that may cause the initiating event. Therefore, this is identified as a gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce A and B Safety Reports. Bruce A and Bruce B designs incorporate engineered barriers and features to prevent failures from errors in operation and maintenance that could result in harmful consequences in terms of redundancy, diversity and separation. However, these provisions were not always explicitly defined as requirements in terms of errors in operation and maintenance. As a general requirement it is not possible to make additional wholesale design changes to prevent the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences. In addition, Operating Policies and Principles and supporting operating documentation and operating and maintenance procedures provide additional barriers to minimize the likelihood of events initiated as a result of operator errors or errors in maintenance.

	Capability of the current design of SSCs resulting from human errors will be analyzed as part of AI 090739 under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. Errors in maintenance are also considered as leading to equipment failures which are covered under PIEs and the event classification. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_6.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF01-01-15
Issue/Gap Description	As discussed in Clause 4.2.1, the DSA in the Safety Report does not distinguish between AOO and DBA and does not address BDBAs

	explicitly. DECs were not considered in the design basis; however, the design basis includes some event sequences that would be categorized as BDBAs. The limits for AOOs are currently taken to be the same as for DBAs (this is the same gap previously identified for Clause 4.2.1). Since the DECs and BDBAs are not explicitly addressed in the design, this is identified as a gap. (Gap)
Rationale	Bruce Power is in compliance with the current licensing basis which is documented in the Bruce A and B Safety Reports. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant. Design Extension Condition (DEC) is a plant state introduced as a subset of BDBAs that have to be considered for the prevention and mitigation of radiation exposures in the plant design. Design requirements associated with such a change affects the plant design as a whole and requires definition of a set of new design limits and acceptance criteria to be established as a pre-requisite. This would require a fundamentally different approach in the design of SSCs and implementation of changes for the prevention and mitigation of radiation hazards associated with DBAs, DECs and BDBAs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_6.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF01-01-16
Issue/Gap Description	As discussed in Clause 4.2.1, the DSA in the Safety Report does not distinguish between AOO and DBA and does not address BDBAs

	explicitly. DECs were not considered in the design basis; however, the design basis includes some event sequences that would be categorized as BDBAs. The limits for AOOs are currently taken to be the same as for DBAs (this is the same gap previously identified for Clause 4.2.1). Since the DECs and BDBAs are not explicitly addressed in the design, this is identified as a gap. (Gap)
Rationale	Bruce Power is in compliance with the current licensing basis which is documented in the Bruce A and B Safety Reports. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant. Design Extension Condition (DEC) is a plant state introduced as a subset of BDBAs that have to be considered for the prevention and mitigation of radiation exposures in the plant design. Design requirements associated with such a change affects the plant design as a whole and requires definition of a set of new design limits and acceptance criteria to be established as a pre-requisite. This would require a fundamentally different approach in the design of SSCs and implementation of changes for the prevention and mitigation of radiation hazards associated with DBAs, DECs and BDBAs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.10_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.10 Safety support system
Requirement Assessed	The safety support systems shall ensure that the fundamental safety functions are available in operational states, DBAs and DECs. Safety support systems provide services such as electrical power, compressed air, water, and air conditioning and ventilation to systems important to safety. Where normal services are provided from external sources, backup safety support systems shall also be available onsite. The design shall incorporate emergency safety support systems to cope with the possibility of loss of normal service and, where applicable, concurrent loss of backup systems. The systems that provide normal services, backup services and emergency services shall have: 1. sufficient capacity to meet the load requirements of the systems that perform the fundamental safety functions 2. availability and reliability commensurate with the systems to which they supply the service The emergency support systems shall: 1. be independent of normal and backup systems 2. support continuity of the fundamental safety functions until long-term (normal or backup) service is re-established: a. without the need for operator action to connect temporary onsite services for at least 8 hours b. without the need for offsite services and support for at least 72 hours 3. have a capacity margin that allows for future increases in demand 4. be testable under design load conditions, where practicable Guidance The design basis for any compressed air system that serves an item important to safety at the NPP should specify the quality, flow rate and cleanness of the air to be provided. Systems for air conditioning, air heating, air cooling and ventilation should be provided (as appropriate) in auxiliary rooms or other areas at the

	nuclear power plant, so as to maintain the required environmental conditions for systems and components important to safety, in all plant states. Pre-installed equipment can be credited for accident mitigation after 30 minutes where only control room actions are needed or after 1 hour if field actions are needed. These actions should be limited to operating valves, starting pumps, etc. Guidance is provided in section 8.10.4 for justification of such actions. If equipment is not pre-installed, but is stored onsite, it can normally be credited after 8 hours. However, this should be justified based on an assessment of the actions required and the availability of procedures and training to support those actions. It is possible that longer times may be necessary for complex actions. Equipment or supplies stored offsite or support staff from offsite should not normally be credited for 72 hours. Again, the value used should be justified and may be longer. Guidance on redundant connection points for temporary services is described in section 7.3.4.1.
Macro-Gap	SF01-09-16
Issue/Gap Description	A summary of the operator actions credited in the safety analysis is documented in Section 1.3 of Part 3 of the Safety Report. The current design documentation does not specifically address the timing requirements introduced in this clause; therefore this is assessed as a gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis which puts operator action times inside the control room at 15 minutes as documented in the Bruce B Safety Report. Classification of all SSCs for determining safety importance based on the time following a PIE at which the SSC will be called upon to operate, and the expected duration of that operation would lead to fundamental design changes both at the plant layout, system and component level which is impracticable. Timing aspects of system operation following a PIE are addressed by the current safety analysis and also in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.13.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.13.1 Seismic design and classification
Requirement Assessed	The design authority shall ensure that seismically qualified SSCs important to safety are qualified to a design-basis earthquake (DBE), and ensure that they are categorized accordingly. This shall apply to: 1. SSCs whose failure could directly or indirectly cause an accident leading to core damage 2. SSCs restricting the release of radioactive material to the environment 3. SSCs that assure the subcriticality of stored nuclear material 4. SSCs such as radioactive waste tanks containing radioactive material that, if released, would exceed regulatory dose limits The design of these SSCs shall also meet the DBE criteria to maintain all essential attributes, such as pressure boundary integrity, leak-tightness, operability, and proper position in the event of a DBE. The design shall ensure that no substantive damage to these SSCs will be caused by the failure of any other SSC under DBE conditions. Seismic fragility levels shall be evaluated for SSCs important to safety by analysis or, where possible, by testing. A beyond-design-basis earthquake (BDBE) shall be identified that meets the requirements for identification of DECs as described in section 7.3.4. SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions. Such demonstration shall provide high confidence of low probability of failure (HCLPF) under BDBE conditions for these SSCs. This demonstration need not be seismic qualification by testing. Guidance The seismic design of an NPP should account for: • technical safety objectives and corresponding load categories • seismic input motion • seismic classification • structural layout criteria • seismic analysis and design of structural systems, subsystems and equipment • seismic testing and instrumentation

	Design and beyond design load categories are defined to demonstrate structural performance in operational states, DBAs and DECs. In addition, beyond design load categories are considered for structural performance in DECs. Earthquake load is not part of the normal load category corresponding to normal operation. Site design earthquake load, according to the CSA N289 series on seismic design and qualification, is defined under the severe load category corresponding to AOO. A DBE is defined as a part of the abnormal or extreme load category corresponding to DBA. BDBE load should be considered under DECs. Seismic input motion, derived from the DBE, should be based on seismicity and geologic conditions at the site and expressed in such a manner that it can be applied for the qualification of SSCs. The DBE is defined by multiplying the mean site specific uniform hazard spectrum with a probability of occurrence of 1E-4/yr by a design factor, defined in the standard ASCE 43-05, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities. The probability of occurrence of the defined DBE is therefore equivalent to the probability of DBAs. A minimum seismic input motion, consistent with national or international standards, should be considered in the design phase for the DBE. The minimum seismic input motion should take into account frequencies of interest for SSCs. Structural layout criteria, including structural separation, should follow best engineering practices and lessons learned from past earthquakes. Modelling of soil-structure interaction (SSI) should be based on geotechnical investigation and taking into account the random nature of soil material properties and inherent uncertainties incorporated in soil constitutive models used in the analysis. To account for uncertainties in soil properties a range with at least three values (upper limit, best estimate and lower limit) should be taken into account in the analysis according to CSA N289.3, Design procedures for seismic qualification of nuclear power plants, clause 5.2.3. The analysis of SSI should take into account all effects due to kinematic interaction (effect of applied seismic ground motion on massless structure) and inertial interaction (inertial forces developed in the structure due to the seismic ground motion). The detail and sophistication of soil-structure models should be in accordance with the purposes of the analyses. The frequency range of interest determines aspects of the structure model and the SSI model parameters. The frequency range of interest should be based on the combination of the frequency range of the earthquake input, the soil properties, the frequency range of building response (including response of subsystems modelled in the main building or structure model), and the frequency range of the response parameter of interest. Refined finite element meshes and increased analytical rigor are required to transmit higher frequencies through the analytical models.

	Damping ratios for structural systems and sub-systems should be taken into account according to recognized standards such as ASCE 43-05 and CSA N289.3. For generating the in-structure response spectra to be used as input to the structure mounted systems and components, Response Level 1 damping of the structure is more appropriate unless the structure response generally exceeds demand over capacity factor given in ASCE 43-05. The seismic design of structural systems should be categorized according to seismic design category (SDC) 1 to 5 as per ASCE 43-05. SDC 1 and 2 structural systems should be in accordance with the National Building Code of Canada, Division B, Part 4. According to the Code, SDC 1 should be as normal and SDC 2 as post-disaster. All structures important to safety are classified as SDC 5. However, the designer may still classify some structures as SDC 3, 4 and 5 provided that they include proper justification. Guidance on SDC 3, 4 and 5 (if SDC 3 and 4 are used) structural systems are provided as follows: • for concrete containment, the design should be based on the American Society of Civil Engineers, ASCE 43-05 (SDC 5, limit state D) and CSA N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants • for steel containment, the design should be based on ASCE 43-05 (SDC 5), 2010 ASME Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NE: Class MC Components and U.S. NRC Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components • for concrete and steel safety related structures the design should be based on ASCE 43-05 (SDC 5, limit state D) and CSA N291, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants For all safety design categories in an NPP, ductility requirements should be in accordance with CSA-A23.3, Design of Concrete Structures for concrete structures and CSA S16, Design of Steel Structures for steel structures assuming that the structures are ductile or type D. These ductility requirements should provide margins for the BDBE. Sub-system analysis should follow the guidance presented for structural systems with the following criteria specific to sub-system supports: • in-structure response spectra • in-structure time response histories The methods of defining in-structure response spectra or in-structure time-histories as well as application of this seismic input to sub-systems and

	components should be in accordance with ASCE 04, Seismic Analysis for Safety-Related Nuclear Structures. Multiple support seismic input of sub-systems and components should take into account their inertial and kinematic components. The analysis should follow ASCE 04 or CSA N289.3, Design procedures for seismic qualification of nuclear power plants. Determination of the number of earthquake cycles for sub-system analysis should be in accordance with U.S. NRC NUREG-0800, Standard Review Plan, section 3.7.3, Seismic Subsystem Analysis as well as seismic analysis of above-ground tanks. Seismic design of sub-systems and components should be in accordance with ASCE 43-05 section 8.2.3 which follows ASME Code. For equipment qualified by testing, multi-axis, multi-frequency testing is acceptable for the DBE in accordance with the requirement of IEEE 344-2004 – IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations and that the testing response spectrum should be at least a factor of 1.4 times the required response spectrum throughout the frequency range. Any deviation from this should be conservatively justified on a case-by-case basis. Any evaluation for BDBE should utilize the methodology in the Electrical Power Research Institute, (EPRI) TR-103959, Methodology for Developing Seismic Fragilities to determine if a HCLPF goal is met. Seismic instrumentation design should follow CSA-N289.5, Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities which itemizes the requirements for single and multiple unit site seismic instrumentation. Beyond-design-basis margin should be such that seismically induced SSC failure probabilities do not contribute to the total core damage frequency and small and large release frequency to the extent that they do not meet the safety goals. To support meeting the safety goals, the acceptance criterion for BDBE should demonstrate that the plant HCLPF is at least 1.67 times the DBE. Assessment and validation of margins for beyond-design-basis earthquakes should be considered, including the metric HCLPF. The seismic isolation of SSCs is an acceptable design approach to limit seismic demand. Seismic isolation devices should be designed, manufactured and installed to withstand a seismic action defined by a DBE without any failure, preserving its mechanical resistance and full load bearing capacity during and after the earthquake. Moreover, the devices and the whole structural system should be designed to withstand a BDBE up to 2 times the spectral accelerations of the DBE without major damage

	and preserving its function. It includes the provisions to accommodate the structural displacements up to 2 times the displacements under DBE conditions.
Macro-Gap	SF01-01-15
Issue/Gap Description	For seismic events and high winds, the SSCs credited to prevent severe core damage are defined by the PSA for these hazards. The assessment identified a number of recommendations, several of which have been dispositioned. The remaining assessment recommendations will be further evaluated via a conceptual engineering process that is currently planned for completion by the end of 2015. Since the assessment is limited to the direct effects of a single initiator and covers the SSCs required to prevent severe core damage, compliance with the new requirement for a BDBE introduced in this clause cannot be confirmed. Therefore, it is assessed as gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis as described in the Bruce A Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. Section 7.13.1 requires definition of a beyond-design-basis earthquake (BDBE) that meets the requirements for identification of DECs. SSCs credited to function during and after a BDBE and demonstration of their capability to perform their intended function under the expected conditions are also required. This would require redesign of the whole of PHT and Moderator Systems, containment system and its structure, penetrations, extension of containment boundary. This is simply not possible for the current plant as such requirements can only be implemented in a new plant design. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.13.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.13.1 Seismic design and classification
Requirement Assessed	The design authority shall ensure that seismically qualified SSCs important to safety are qualified to a design-basis earthquake (DBE), and ensure that they are categorized accordingly. This shall apply to: 1. SSCs whose failure could directly or indirectly cause an accident leading to core damage 2. SSCs restricting the release of radioactive material to the environment 3. SSCs that assure the subcriticality of stored nuclear material 4. SSCs such as radioactive waste tanks containing radioactive material that, if released, would exceed regulatory dose limits The design of these SSCs shall also meet the DBE criteria to maintain all essential attributes, such as pressure boundary integrity, leak-tightness, operability, and proper position in the event of a DBE. The design shall ensure that no substantive damage to these SSCs will be caused by the failure of any other SSC under DBE conditions. Seismic fragility levels shall be evaluated for SSCs important to safety by analysis or, where possible, by testing. A beyond-design-basis earthquake (BDBE) shall be identified that meets the requirements for identification of DECs as described in section 7.3.4. SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions. Such demonstration shall provide high confidence of low probability of failure (HCLPF) under BDBE conditions for these SSCs. This demonstration need not be seismic qualification by testing. Guidance The seismic design of an NPP should account for: • technical safety objectives and corresponding load categories • seismic input motion • seismic classification • structural layout criteria • seismic analysis and design of structural systems, subsystems and equipment • seismic testing and instrumentation

	Design and beyond design load categories are defined to demonstrate structural performance in operational states, DBAs and DECs. In addition, beyond design load categories are considered for structural performance in DECs. Earthquake load is not part of the normal load category corresponding to normal operation. Site design earthquake load, according to the CSA N289 series on seismic design and qualification, is defined under the severe load category corresponding to AOO. A DBE is defined as a part of the abnormal or extreme load category corresponding to DBA. BDBE load should be considered under DECs. Seismic input motion, derived from the DBE, should be based on seismicity and geologic conditions at the site and expressed in such a manner that it can be applied for the qualification of SSCs. The DBE is defined by multiplying the mean site specific uniform hazard spectrum with a probability of occurrence of 1E-4/yr by a design factor, defined in the standard ASCE 43-05, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities. The probability of occurrence of the defined DBE is therefore equivalent to the probability of DBAs. A minimum seismic input motion, consistent with national or international standards, should be considered in the design phase for the DBE. The minimum seismic input motion should take into account frequencies of interest for SSCs. Structural layout criteria, including structural separation, should follow best engineering practices and lessons learned from past earthquakes. Modelling of soil-structure interaction (SSI) should be based on geotechnical investigation and taking into account the random nature of soil material properties and inherent uncertainties incorporated in soil constitutive models used in the analysis. To account for uncertainties in soil properties a range with at least three values (upper limit, best estimate and lower limit) should be taken into account in the analysis according to CSA N289.3, Design procedures for seismic qualification of nuclear power plants, clause 5.2.3. The analysis of SSI should take into account all effects due to kinematic interaction (effect of applied seismic ground motion on massless structure) and inertial interaction (inertial forces developed in the structure due to the seismic ground motion). The detail and sophistication of soil-structure models should be in accordance with the purposes of the analyses. The frequency range of interest determines aspects of the structure model and the SSI model parameters. The frequency range of interest should be based on the combination of the frequency range of the earthquake input, the soil properties, the frequency range of building response (including response of subsystems modelled in the main building or structure model), and the frequency range of the response parameter of interest. Refined finite element meshes and increased analytical rigor are required to transmit higher frequencies through the analytical models.

	Damping ratios for structural systems and sub-systems should be taken into account according to recognized standards such as ASCE 43-05 and CSA N289.3. For generating the in-structure response spectra to be used as input to the structure mounted systems and components, Response Level 1 damping of the structure is more appropriate unless the structure response generally exceeds demand over capacity factor given in ASCE 43-05. The seismic design of structural systems should be categorized according to seismic design category (SDC) 1 to 5 as per ASCE 43-05. SDC 1 and 2 structural systems should be in accordance with the National Building Code of Canada, Division B, Part 4. According to the Code, SDC 1 should be as normal and SDC 2 as post-disaster. All structures important to safety are classified as SDC 5. However, the designer may still classify some structures as SDC 3, 4 and 5 provided that they include proper justification. Guidance on SDC 3, 4 and 5 (if SDC 3 and 4 are used) structural systems are provided as follows: • for concrete containment, the design should be based on the American Society of Civil Engineers, ASCE 43-05 (SDC 5, limit state D) and CSA N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants • for steel containment, the design should be based on ASCE 43-05 (SDC 5), 2010 ASME Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NE: Class MC Components and U.S. NRC Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components • for concrete and steel safety related structures the design should be based on ASCE 43-05 (SDC 5, limit state D) and CSA N291, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants For all safety design categories in an NPP, ductility requirements should be in accordance with CSA-A23.3, Design of Concrete Structures for concrete structures and CSA S16, Design of Steel Structures for steel structures assuming that the structures are ductile or type D. These ductility requirements should provide margins for the BDBE. Sub-system analysis should follow the guidance presented for structural systems with the following criteria specific to sub-system supports: • in-structure response spectra • in-structure time response histories The methods of defining in-structure response spectra or in-structure time-histories as well as application of this seismic input to sub-systems and

	components should be in accordance with ASCE 04, Seismic Analysis for Safety-Related Nuclear Structures. Multiple support seismic input of sub-systems and components should take into account their inertial and kinematic components. The analysis should follow ASCE 04 or CSA N289.3, Design procedures for seismic qualification of nuclear power plants. Determination of the number of earthquake cycles for sub-system analysis should be in accordance with U.S. NRC NUREG-0800, Standard Review Plan, section 3.7.3, Seismic Subsystem Analysis as well as seismic analysis of above-ground tanks. Seismic design of sub-systems and components should be in accordance with ASCE 43-05 section 8.2.3 which follows ASME Code. For equipment qualified by testing, multi-axis, multi-frequency testing is acceptable for the DBE in accordance with the requirement of IEEE 344-2004 – IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations and that the testing response spectrum should be at least a factor of 1.4 times the required response spectrum throughout the frequency range. Any deviation from this should be conservatively justified on a case-by-case basis. Any evaluation for BDBE should utilize the methodology in the Electrical Power Research Institute, (EPRI) TR-103959, Methodology for Developing Seismic Fragilities to determine if a HCLPF goal is met. Seismic instrumentation design should follow CSA-N289.5, Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities which itemizes the requirements for single and multiple unit site seismic instrumentation. Beyond-design-basis margin should be such that seismically induced SSC failure probabilities do not contribute to the total core damage frequency and small and large release frequency to the extent that they do not meet the safety goals. To support meeting the safety goals, the acceptance criterion for BDBE should demonstrate that the plant HCLPF is at least 1.67 times the DBE. Assessment and validation of margins for beyond-design-basis earthquakes should be considered, including the metric HCLPF. The seismic isolation of SSCs is an acceptable design approach to limit seismic demand. Seismic isolation devices should be designed, manufactured and installed to withstand a seismic action defined by a DBE without any failure, preserving its mechanical resistance and full load bearing capacity during and after the earthquake. Moreover, the devices and the whole structural system should be designed to withstand a BDBE up to 2 times the spectral accelerations of the DBE without major damage

	and preserving its function. It includes the provisions to accommodate the structural displacements up to 2 times the displacements under DBE conditions.
Macro-Gap	SF01-16-16
Issue/Gap Description	The Design Basis Earthquake (DBE) for Bruce B is expressed in the form of a generic response spectra (90th percentile) derived from a study of response spectra recorded of large earthquakes and normalized to a site-specific peak ground acceleration. The peak ground acceleration of 0.05g was selected to correspond to an occurrence rate of less than 1E-3 per year. The current peak ground acceleration does not appear to satisfy guidance provided in this clause (7.13.1 of REGDOC-2.5.2), which defines a Design Basis Earthquake (DBE) with a "probability of occurrence of 1E-4/y by a design factor defined in the standard ASCE 43-05". This DBE definition is consistent with CSA N289.3-10, clause 3.1 which states the DBE states "an engineering representation of potentially severe effects at the site due to earthquake ground motions having selected probability of exceedance of 1E-4/y or such a probability level as is acceptable to the authority having jurisdiction". To further this, the minimum design ground response spectra is defined in CSA N289.3-10 clause 4.2 which states the standard-shape ground response spectrum anchored to a peak ground acceleration of 0.1g on rock. Bruce Power has received a formal interpretation of Clause 4.2 of CSA Standard N289.3-10 which states that the intent of the clause is applicable only to the design of SSCs of new nuclear power plants [NK29-CORR-00531-12453]. Therefore this is considered a gap against the guidance of this clause (Gap 1).
Rationale	The design basis earthquake for Bruce B is peak ground acceleration of 0.05g was selected to correspond to an occurrence rate of less than 1E-3 per year. It is not practicable to make wholesale design changes to comply with the intent of this clause which impacts all SSCs important to safety which drives all structural and component level design requirements that is applicable to new nuclear plants as it can only be implemented at the initial design stage. It should also be noted that Bruce Power has received a formal interpretation of Clause 4.2 of CSA Standard N289.3-10 which states that the intent of the clause is applicable only to the design of SSCs of new nuclear power plants [NK29-CORR-00531-12453]. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.13.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.13.1 Seismic design and classification
Requirement Assessed	The design authority shall ensure that seismically qualified SSCs important to safety are qualified to a design-basis earthquake (DBE), and ensure that they are categorized accordingly. This shall apply to: 1. SSCs whose failure could directly or indirectly cause an accident leading to core damage 2. SSCs restricting the release of radioactive material to the environment 3. SSCs that assure the subcriticality of stored nuclear material 4. SSCs such as radioactive waste tanks containing radioactive material that, if released, would exceed regulatory dose limits The design of these SSCs shall also meet the DBE criteria to maintain all essential attributes, such as pressure boundary integrity, leak-tightness, operability, and proper position in the event of a DBE. The design shall ensure that no substantive damage to these SSCs will be caused by the failure of any other SSC under DBE conditions. Seismic fragility levels shall be evaluated for SSCs important to safety by analysis or, where possible, by testing. A beyond-design-basis earthquake (BDBE) shall be identified that meets the requirements for identification of DECs as described in section 7.3.4. SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions. Such demonstration shall provide high confidence of low probability of failure (HCLPF) under BDBE conditions for these SSCs. This demonstration need not be seismic qualification by testing. Guidance The seismic design of an NPP should account for: • technical safety objectives and corresponding load categories • seismic input motion • seismic classification • structural layout criteria • seismic analysis and design of structural systems, subsystems and equipment • seismic testing and instrumentation

	Design and beyond design load categories are defined to demonstrate structural performance in operational states, DBAs and DECs. In addition, beyond design load categories are considered for structural performance in DECs. Earthquake load is not part of the normal load category corresponding to normal operation. Site design earthquake load, according to the CSA N289 series on seismic design and qualification, is defined under the severe load category corresponding to AOO. A DBE is defined as a part of the abnormal or extreme load category corresponding to DBA. BDBE load should be considered under DECs. Seismic input motion, derived from the DBE, should be based on seismicity and geologic conditions at the site and expressed in such a manner that it can be applied for the qualification of SSCs. The DBE is defined by multiplying the mean site specific uniform hazard spectrum with a probability of occurrence of 1E-4/yr by a design factor, defined in the standard ASCE 43-05, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities. The probability of occurrence of the defined DBE is therefore equivalent to the probability of DBAs. A minimum seismic input motion, consistent with national or international standards, should be considered in the design phase for the DBE. The minimum seismic input motion should take into account frequencies of interest for SSCs. Structural layout criteria, including structural separation, should follow best engineering practices and lessons learned from past earthquakes. Modelling of soil-structure interaction (SSI) should be based on geotechnical investigation and taking into account the random nature of soil material properties and inherent uncertainties incorporated in soil constitutive models used in the analysis. To account for uncertainties in soil properties a range with at least three values (upper limit, best estimate and lower limit) should be taken into account in the analysis according to CSA N289.3, Design procedures for seismic qualification of nuclear power plants, clause 5.2.3. The analysis of SSI should take into account all effects due to kinematic interaction (effect of applied seismic ground motion on massless structure) and inertial interaction (inertial forces developed in the structure due to the seismic ground motion). The detail and sophistication of soil-structure models should be in accordance with the purposes of the analyses. The frequency range of interest determines aspects of the structure model and the SSI model parameters. The frequency range of interest should be based on the combination of the frequency range of the earthquake input, the soil properties, the frequency range of building response (including response of subsystems modelled in the main building or structure model), and the frequency range of the response parameter of interest. Refined finite element meshes and increased analytical rigor are required to transmit higher frequencies through the analytical models.

Damping ratios for structural systems and sub-systems should be taken into account according to recognized standards such as ASCE 43-05 and CSA N289.3. For generating the in-structure response spectra to be used as input to the structure mounted systems and components, Response Level 1 damping of the structure is more appropriate unless the structure response generally exceeds demand over capacity factor given in ASCE 43-05.

The seismic design of structural systems should be categorized according to seismic design category (SDC) 1 to 5 as per ASCE 43-05.

SDC 1 and 2 structural systems should be in accordance with the National Building Code of Canada, Division B, Part 4. According to the Code, SDC 1 should be as normal and SDC 2 as post-disaster.

All structures important to safety are classified as SDC 5. However, the designer may still classify some structures as SDC 3, 4 and 5 provided that they include proper justification. Guidance on SDC 3, 4 and 5 (if SDC 3 and 4 are used) structural systems are provided as follows:

- for concrete containment, the design should be based on the American Society of Civil Engineers, ASCE 43-05 (SDC 5, limit state D) and CSA N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants
- for steel containment, the design should be based on ASCE 43-05 (SDC 5), 2010 ASME Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NE: Class MC Components and U.S. NRC Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components
- for concrete and steel safety related structures the design should be based on ASCE 43-05 (SDC 5, limit state D) and CSA N291, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants

For all safety design categories in an NPP, ductility requirements should be in accordance with CSA-A23.3, Design of Concrete Structures for concrete structures and CSA S16, Design of Steel Structures for steel structures assuming that the structures are ductile or type D. These ductility requirements should provide margins for the BDBE.

Sub-system analysis should follow the guidance presented for structural systems with the following criteria specific to sub-system supports:

- in-structure response spectra
- in-structure time response histories

The methods of defining in-structure response spectra or in-structure time-histories as well as application of this seismic input to sub-systems and

	components should be in accordance with ASCE 04, Seismic Analysis for Safety-Related Nuclear Structures. Multiple support seismic input of sub-systems and components should take into account their inertial and kinematic components. The analysis should follow ASCE 04 or CSA N289.3, Design procedures for seismic qualification of nuclear power plants. Determination of the number of earthquake cycles for sub-system analysis should be in accordance with U.S. NRC NUREG-0800, Standard Review Plan, section 3.7.3, Seismic Subsystem Analysis as well as seismic analysis of above-ground tanks. Seismic design of sub-systems and components should be in accordance with ASCE 43-05 section 8.2.3 which follows ASME Code. For equipment qualified by testing, multi-axis, multi-frequency testing is acceptable for the DBE in accordance with the requirement of IEEE 344-2004 – IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations and that the testing response spectrum should be at least a factor of 1.4 times the required response spectrum throughout the frequency range. Any deviation from this should be conservatively justified on a case-by-case basis. Any evaluation for BDBE should utilize the methodology in the Electrical Power Research Institute, (EPRI) TR-103959, Methodology for Developing Seismic Fragilities to determine if a HCLPF goal is met. Seismic instrumentation design should follow CSA-N289.5, Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities which itemizes the requirements for single and multiple unit site seismic instrumentation. Beyond-design-basis margin should be such that seismically induced SSC failure probabilities do not contribute to the total core damage frequency and small and large release frequency to the extent that they do not meet the safety goals. To support meeting the safety goals, the acceptance criterion for BDBE should demonstrate that the plant HCLPF is at least 1.67 times the DBE. Assessment and validation of margins for beyond-design-basis earthquakes should be considered, including the metric HCLPF. The seismic isolation of SSCs is an acceptable design approach to limit seismic demand. Seismic isolation devices should be designed, manufactured and installed to withstand a seismic action defined by a DBE without any failure, preserving its mechanical resistance and full load bearing capacity during and after the earthquake. Moreover, the devices and the whole structural system should be designed to withstand a BDBE up to 2 times the spectral accelerations of the DBE without major damage

	and preserving its function. It includes the provisions to accommodate the structural displacements up to 2 times the displacements under DBE conditions.
Macro-Gap	SF01-01-16
Issue/Gap Description	A Probabilistic Seismic Hazard Assessment was done for the Bruce B site in 2011 [NK29-03500.8 P NSAS, Rev.1] which does provide information about earthquakes beyond the DBE level. The Seismic Hazard Assessment does not however identify a BDBE or provide assessment. Compliance with the new requirement for a BDBE introduced in this clause cannot be confirmed. Therefore, it is assessed as gap (Gap2). Although it is noted, in accordance with clause 5.2.4.2 of CSA N289.3 to evaluate beyond design basis events as being applicable to new plants, not existing plants.
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. Section 7.13.1 requires definition of a beyond-design-basis earthquake (BDBE) that meets the requirements for identification of DECs. SSCs credited to function during and after a BDBE and demonstration of their capability to perform their intended function under the expected conditions are also required. This would require redesign of the whole of PHT and Moderator Systems, containment system and its structure, penetrations, extension of containment boundary. This is simply not possible for the current plant as such requirements can only be implemented in a new plant design. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.15.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.15.1 Design
Requirement Assessed	The NPP design shall specify the required performance for the safety functions of the civil structures in operational states, DBAs and DECs. Civil structures important to safety shall be designed and located so as to minimize the probabilities and effects of internal hazards such as fire, explosion, smoke, flooding, missile generation, pipe whip, jet impact, or release of fluid due to pipe breaks. External hazards such as earthquakes, floods, high winds, tornadoes, tsunamis, and extreme meteorological conditions shall be considered in the design of civil structures. Settlement analysis and evaluation of soil capacity shall include consideration of the effects of fluctuating ground water on the foundations, and identification and evaluation of potential liquefiable soil strata and slope failure. Civil structures important to safety shall be designed to meet the serviceability, strength, and stability requirements for all possible load combinations under the categories of normal operation, AOO, DBA and DEC conditions, including external hazards. The serviceability considerations shall include, without being limited to, deflection, vibration, permanent deformation, cracking, and settlement. The design specifications shall also define all loads and load combinations, with due consideration given to the probability of concurrence and loading time history. Environmental effects shall be considered in the design of civil structures and the selection of construction materials. The choice of construction material shall be commensurate with the designed service life and potential life extension of the plant. The plant safety assessment shall include structural analyses for all civil structures important to safety. Guidance The design authority should provide the design principles, design basis requirements and criteria, and applicable codes and standards, design and analysis procedures, the assumed boundary conditions and the computer codes used in the analysis and design. All internal and external hazard loads are specified in section 7.4.

	Earthquake design input loads and impacts of malevolent acts, including large aircraft crash can be found in sections 7.13 and 7.22, respectively. Load categories corresponding to the plant states are defined in this section so as to demonstrate structural performances as follows: • normal condition loads which are expected during the assumed design life of the NPP • AOO loads (or severe environmental loads) • DBA loads (or abnormal or extreme environmental loads) • DEC loads (or beyond-design loads) The design should identify all DEC loads considered in the structure design and provide the assessment methodology and acceptance criteria. The structural design should withstand, accommodate or avoid foundation settlement (total and differential), according to its performance requirements. The structural design should consider the impact of aging on the structure and its material. The design should include sufficient safety margins for the buildings and structures that are important to safety. The physical and material description of each civil structure and its base slab should include: • the type of structure, and its structural and functional characteristics • the geometry of the structures, including sketches showing plan views at various elevations and sections (at least two orthogonal directions) • the relationship between adjacent structures, including any separation or structural ties • the type of base slab and its arrangement with the methods of transferring horizontal shears (such as those seismically induced) to the foundation media Containment structure The design should specify the safety requirements for the containment building or system, including, for example, its structural strength, leak tightness, and resistance to steady-state and transient loads (such as those arising from pressure, temperature, radiation, and mechanical impact) that could be caused by postulated internal and external hazards. In addition, the design should specify the safety requirements and design features for the containment internal structures, (such as the reactor vault structure, the shielding doors, the airlocks, and the access control and facilities). The design of the containment structure should include:

	• base slab and sub-base • containment wall and dome design • containment wall openings and penetrations • pre-stressing system • containment liner and its attachment method The design pressure of the containment building should be determined by increasing by at least 10% the peak pressure that would be generated by the DBA (refer to clause 4.49 of IAEA NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants). Ultimate internal pressure capacity should be provided for the containment building structures including containment penetrations. If the containment building foundation is a common mat slab which is not separated from the other buildings foundation, the impact should be evaluated. Concrete containment structures should be designed and constructed in accordance with the CSA N287 series, as applicable: • N287.1, General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, for general requirements in documentation of design specification and design reports • N287.2, Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, for material • N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for design • N287.4, Construction, Fabrication and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, and N287.5, Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants, for containment construction and inspection • N287.6, Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants, for pressure test before operation Steel containment structures should be designed according to the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, Class MC Components or equivalent standard. Stability of the containment vessel and appurtenances should be evaluated using ASME Code Case N-284-1, Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC. For other requirements on the design of containment structures, refer to section 8.6.2 of this regulatory document. Safety-related structures

	The safety-related structures other than the containment should be designed and constructed in accordance with CSA N291, Requirements for safety-related structures for CANDU nuclear power plants. The design of other safety-related structures should include: • internal structures of reactor building • service (auxiliary) building • fuel storage building • control building • diesel generator building • containment shield building, if applicable • other safety-related structures defined by the design • turbine building (for boiling water reactor)
Macro-Gap	SF01-03-15
Issue/Gap Description	The design requirement for civil structures important to safety to meet the serviceability, strength and stability requirements for all possible load combinations under operational states and DBAs is extended to include DECs (new requirement). (Gap) As discussed in Part 2, Section 7.4.1 of the Safety Report, Bruce A design does not meet these requirements, as documented in [NK21-CORR-00531-11005]. Specifically, internal hazards were not a primary consideration for the original Bruce A design and layout of civil structures important to safety. As a result, the requirements associated with internal hazards such as pipe whip and jet impingement were not fully addressed. As discussed in supporting documentation for NK21-CORR-00531-11567 CNSC has accepted the results of the Pipe Whip and Jet Impingement Assessment of Piping Inside the Reactor Vault. The results of the assessment concluded that no design changes are required in the Units 1 and 2 vaults as a result of pipe-whip or jet impingement.
Rationale	Bruce A plant design basis, documented in the Bruce A Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not practicable to make wholesale design changes to Civil Structures to protect against internal events and specifically those associated with DECs. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. As those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-003 Assess pipe whip and jet impingement- (Pipe Whip and Jet Impingement Assessment of Piping Inside the Reactor Vault complete for

	Bruce A and results have been accepted by the CNSC) GIO-019 Assess and improve seismic margins GIO-087 Bruce A Fire Protection Upgrades to Align with CSA-N293-14 (complete) It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.15.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.15.1 Design
Requirement Assessed	The NPP design shall specify the required performance for the safety functions of the civil structures in operational states, DBAs and DECs. Civil structures important to safety shall be designed and located so as to minimize the probabilities and effects of internal hazards such as fire, explosion, smoke, flooding, missile generation, pipe whip, jet impact, or release of fluid due to pipe breaks. External hazards such as earthquakes, floods, high winds, tornadoes, tsunamis, and extreme meteorological conditions shall be considered in the design of civil structures. Settlement analysis and evaluation of soil capacity shall include consideration of the effects of fluctuating ground water on the foundations, and identification and evaluation of potential liquefiable soil strata and slope failure. Civil structures important to safety shall be designed to meet the serviceability, strength, and stability requirements for all possible load combinations under the categories of normal operation, AOO, DBA and DEC conditions, including external hazards. The serviceability considerations shall include, without being limited to, deflection, vibration, permanent deformation, cracking, and settlement. The design specifications shall also define all loads and load combinations, with due consideration given to the probability of concurrence and loading time history. Environmental effects shall be considered in the design of civil structures and the selection of construction materials. The choice of construction material shall be commensurate with the designed service life and potential life extension of the plant. The plant safety assessment shall include structural analyses for all civil structures important to safety. Guidance The design authority should provide the design principles, design basis requirements and criteria, and applicable codes and standards, design and analysis procedures, the assumed boundary conditions and the computer codes used in the analysis and design. All internal and external hazard loads are specified in section 7.4.

	Earthquake design input loads and impacts of malevolent acts, including large aircraft crash can be found in sections 7.13 and 7.22, respectively. Load categories corresponding to the plant states are defined in this section so as to demonstrate structural performances as follows: • normal condition loads which are expected during the assumed design life of the NPP • AOO loads (or severe environmental loads) • DBA loads (or abnormal or extreme environmental loads) • DEC loads (or beyond-design loads) The design should identify all DEC loads considered in the structure design and provide the assessment methodology and acceptance criteria. The structural design should withstand, accommodate or avoid foundation settlement (total and differential), according to its performance requirements. The structural design should consider the impact of aging on the structure and its material. The design should include sufficient safety margins for the buildings and structures that are important to safety. The physical and material description of each civil structure and its base slab should include: • the type of structure, and its structural and functional characteristics • the geometry of the structures, including sketches showing plan views at various elevations and sections (at least two orthogonal directions) • the relationship between adjacent structures, including any separation or structural ties • the type of base slab and its arrangement with the methods of transferring horizontal shears (such as those seismically induced) to the foundation media Containment structure The design should specify the safety requirements for the containment building or system, including, for example, its structural strength, leak tightness, and resistance to steady-state and transient loads (such as those arising from pressure, temperature, radiation, and mechanical impact) that could be caused by postulated internal and external hazards. In addition, the design should specify the safety requirements and design features for the containment internal structures, (such as the reactor vault structure, the shielding doors, the airlocks, and the access control and facilities). The design of the containment structure should include:

	• base slab and sub-base • containment wall and dome design • containment wall openings and penetrations • pre-stressing system • containment liner and its attachment method The design pressure of the containment building should be determined by increasing by at least 10% the peak pressure that would be generated by the DBA (refer to clause 4.49 of IAEA NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants). Ultimate internal pressure capacity should be provided for the containment building structures including containment penetrations. If the containment building foundation is a common mat slab which is not separated from the other buildings foundation, the impact should be evaluated. Concrete containment structures should be designed and constructed in accordance with the CSA N287 series, as applicable: • N287.1, General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, for general requirements in documentation of design specification and design reports • N287.2, Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, for material • N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for design • N287.4, Construction, Fabrication and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, and N287.5, Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants, for containment construction and inspection • N287.6, Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants, for pressure test before operation Steel containment structures should be designed according to the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, Class MC Components or equivalent standard. Stability of the containment vessel and appurtenances should be evaluated using ASME Code Case N-284-1, Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC. For other requirements on the design of containment structures, refer to section 8.6.2 of this regulatory document. Safety-related structures

	The safety-related structures other than the containment should be designed and constructed in accordance with CSA N291, Requirements for safety-related structures for CANDU nuclear power plants. The design of other safety-related structures should include: • internal structures of reactor building • service (auxiliary) building • fuel storage building • control building • diesel generator building • containment shield building, if applicable • other safety-related structures defined by the design • turbine building (for boiling water reactor)
Macro-Gap	SF01-01-16
Issue/Gap Description	As discussed in section 7.4.1, the current design documentation does not consider internal events as leading to AOOs, DBAs and DECs. This includes the design documentation for civil structures, and is therefore considered a gap (Gap).
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not practicable to make wholesale design changes to Civil Structures to protect against internal events and specifically those associated with DECs. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. as those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-003 Assess pipe whip and jet impingement- (Pipe Whip and Jet Impingement Assessment of Piping Inside the Reactor Vault complete for Bruce A and results have been accepted by the CNSC) GIO-019 Assess and improve seismic margin GIO-092 Bruce B Fire Protection Upgrades to Align with CSA-N293-14 It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.15.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.15.3 Lifting and handling of large loads
Requirement Assessed	The lifting and handling of large and heavy loads, particularly those containing radioactive material, shall be considered in the NPP design. This shall include identification of the large loads, traversing routes and situations where they need to be lifted over areas of the plant that are critical to safety. The design of all cranes and lifting devices shall, therefore, incorporate large margins, appropriate interlocks, and other safety features to accommodate the lifting of large loads. The drop of large loads lifted and handled in areas where there are systems and components that are important to safety shall be taken into account in the design. The potential load due to the large load drop shall be taken into account in the analysis of DBAs.
Macro-Gap	SF01-10-15
Issue/Gap Description	A review of the same clause in a draft version of the RD-337 indicated that the design of Bruce A recognizes the need for lifting heavy loads in a variety of locations and suitable cranes have been installed to perform these lifts. However identification of traversing routes together with justification for safety is not available in the design documentation. Therefore, it is assessed as a gap. (Gap 1)
Rationale	Traversing routes for heavy loads were not explicitly documented in the original design. Bruce Power deals with handling of large and heavy loads as an operational safety issue on an as required basis. In cases such as IPTEs (Infrequently Performed Technical Evolution), specific operational measures and procedures are supported by engineering and safety assessments and implemented with the requisite approvals. One recent example is the generator swap between Units 4 and 2 during the refurbishment outage of Unit 2. Such a requirement is considered to be specifically applicable to a new nuclear plant. Fundamental design changes to the structures are impracticable as such changes will also impact the systems and components associated with the structure. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.15.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.15.3 Lifting and handling of large loads
Requirement Assessed	The lifting and handling of large and heavy loads, particularly those containing radioactive material, shall be considered in the NPP design. This shall include identification of the large loads, traversing routes and situations where they need to be lifted over areas of the plant that are critical to safety. The design of all cranes and lifting devices shall, therefore, incorporate large margins, appropriate interlocks, and other safety features to accommodate the lifting of large loads. The drop of large loads lifted and handled in areas where there are systems and components that are important to safety shall be taken into account in the design. The potential load due to the large load drop shall be taken into account in the analysis of DBAs.
Macro-Gap	SF01-10-16
Issue/Gap Description	The design of Bruce A and B recognizes the need for lifting heavy loads in a variety of locations and suitable cranes have been installed to perform these lifts. However identification of traversing routes together with justification for safety is not available in the design documentation. Therefore, it is assessed as a gap. (Gap 1)
Rationale	Traversing routes for heavy loads were not explicitly documented in the original design. Bruce Power deals with handling of large and heavy loads as an operational safety issue on an as required basis. In cases such as IPTEs (Infrequently Performed Technical Evolution), specific operational measures and procedures are supported by engineering and safety assessments and implemented with the requisite approvals. One recent example is the generator swap between Units 4 and 2 during the refurbishment outage of Unit 2. Such a requirement is considered to be specifically applicable to a new nuclear plant. Fundamental design changes to the structures are impracticable as such changes will also impact the systems and components associated with the structure. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.3.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.3.2 Anticipated operational occurrences
Requirement Assessed	The design shall include provisions such that releases to the public following an AOO do not exceed the dose acceptance criterion provided in section 4.2.1. The design shall also provide that, to the extent practicable, SSCs not involved in the initiation of an AOO shall remain operable following the AOO. The response of the plant to a wide range of AOOs shall allow safe operation or shutdown, if necessary, without the need to invoke provisions beyond Level 1 defence in depth or, at most, Level 2. The facility layout shall be such that equipment is placed at the most suitable location to ensure its immediate availability when operator intervention is required, allowing for safe and timely access during an AOO. Guidance The guidance in this subsection also covers elements common to AOO and DBA. In accordance with the requirements of section 4.3.1 of this regulatory document for Level 2 and Level 3 defence in depth, the design should include the results of the analyses of AOOs and DBAs in order to provide a demonstration of the robustness of the fault tolerance in the engineering design and the effectiveness of the safety systems. The analysis should cover the full range of events over the full range of reactor power. The analysis should also cover all normal operating configurations, including low-power and shutdown states. For a wide range of AOOs, the design should be such that any deviations from normal operation can be detected, and that the control systems can be expected to return the plant to a safe state, normally without the activation of safety systems. For both AOOs and DBAs, there should be high confidence that qualified systems (as identified in REGDOC-2.4.1, Deterministic Safety Analysis) can mitigate the event even when acting alone. In the analysis of AOOs and DBAs for each group of PIEs, it may be sufficient to analyze only a limited number of bounding initiating events, which can represent a bounding response for a group of events. The rationale for the choice of these selected bounding events should be provided. The plant parameters that are important to the outcome of the safety analysis should also be identified. These parameters would typically

	include: • reactor power and its distribution • core component temperatures • fuel cladding oxidation, and deformation • pressures in the primary and secondary systems • containment parameters • temperatures and flows • reactivity coefficients • reactor kinetics parameters • reactivity worth of reactivity devices Those characteristics of the safety systems, including the operating conditions in which the systems are actuated, the time delays, and the systems' capacity after the actuation claimed in the design, should be specified and demonstrated to be consistent with the overall functional and performance requirements of the systems.
Macro-Gap	SF01-01-15
Issue/Gap Description	The requirement for the reactor to be able to continue operation after an AOO basically means that there should be no fuel failure following the event. For several of the AOO cases at Bruce A, this would be the case, e.g., loss of control system functions. For some of the other scenarios, e.g., PHT pump seal failure, the public doses arise from incipient iodine in the HT system and from tritium in the D2O. Thus, repair of the seal would enable the reactor to continue operation. The doses from this event, as calculated in the Bruce A Safety Report using very conservative assumptions are within the currently allowable single failure criterion, but would be outside the AOO limit proposed in Clause 4.2.1 of CNSC REGDOC 2.5.2. Therefore, this is assessed as a gap. (Gap)
Rationale	Bruce A and Bruce B design criteria in terms of event classification and dose limits were different than those specified in REGDOC-2.5.2. However, SSCs were designed for several cases that would be classified as AOOs today. Doses from such events, as calculated in the Bruce A and Bruce B Safety Reports using very conservative assumptions are within the current PROL limits. Such a requirement is considered to be specifically applicable to a new nuclear plant. REGDOC-2.5.2 dose limits for AOOs are an order of magnitude more restrictive as compared to current PROL limits. This would require all SSCs with potential radioactive release risks to be redesigned or would require fundamental changes to fuel design which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.3.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.3.2 Anticipated operational occurrences
Requirement Assessed	The design shall include provisions such that releases to the public following an AOO do not exceed the dose acceptance criterion provided in section 4.2.1. The design shall also provide that, to the extent practicable, SSCs not involved in the initiation of an AOO shall remain operable following the AOO. The response of the plant to a wide range of AOOs shall allow safe operation or shutdown, if necessary, without the need to invoke provisions beyond Level 1 defence in depth or, at most, Level 2. The facility layout shall be such that equipment is placed at the most suitable location to ensure its immediate availability when operator intervention is required, allowing for safe and timely access during an AOO. Guidance The guidance in this subsection also covers elements common to AOO and DBA. In accordance with the requirements of section 4.3.1 of this regulatory document for Level 2 and Level 3 defence in depth, the design should include the results of the analyses of AOOs and DBAs in order to provide a demonstration of the robustness of the fault tolerance in the engineering design and the effectiveness of the safety systems. The analysis should cover the full range of events over the full range of reactor power. The analysis should also cover all normal operating configurations, including low-power and shutdown states. For a wide range of AOOs, the design should be such that any deviations from normal operation can be detected, and that the control systems can be expected to return the plant to a safe state, normally without the activation of safety systems. For both AOOs and DBAs, there should be high confidence that qualified systems (as identified in REGDOC-2.4.1, Deterministic Safety Analysis) can mitigate the event even when acting alone. In the analysis of AOOs and DBAs for each group of PIEs, it may be sufficient to analyze only a limited number of bounding initiating events, which can represent a bounding response for a group of events. The rationale for the choice of these selected bounding events should be provided. The plant parameters that are important to the outcome of the safety analysis should also be identified. These parameters would typically

	include: • reactor power and its distribution • core component temperatures • fuel cladding oxidation, and deformation • pressures in the primary and secondary systems • containment parameters • temperatures and flows • reactivity coefficients • reactor kinetics parameters • reactivity worth of reactivity devices Those characteristics of the safety systems, including the operating conditions in which the systems are actuated, the time delays, and the systems' capacity after the actuation claimed in the design, should be specified and demonstrated to be consistent with the overall functional and performance requirements of the systems.
Macro-Gap	SF01-01-16
Issue/Gap Description	The requirement for the reactor to be able to continue operation after an AOO basically means that there should be no fuel failure following the event. For several of the AOO cases at Bruce B, this would be the case, e.g., loss of control system functions. For some of the other scenarios, e.g., PHT pump seal failure, the public doses arise from incipient iodine in the HT system and from tritium in the D2O. Thus, repair of the seal would enable the reactor to continue operation. The doses from this event, as calculated in the Bruce B Safety Report using very conservative assumptions are within the currently allowable single failure criterion, but would be outside the AOO limit proposed in Clause 4.2.1 of CNSC REGDOC 2.5.2. Therefore, this is assessed as a gap. (Gap)
Rationale	Bruce A and Bruce B design criteria in terms of event classification and dose limits were different than those specified in REGDOC-2.5.2. However, SSCs were designed for several cases that would be classified as AOOs today. Doses from such events, as calculated in the Bruce A and Bruce B Safety Reports using very conservative assumptions are within the current PROL limits. Such a requirement is considered to be specifically applicable to a new nuclear plant. REGDOC-2.5.2 dose limits for AOOs are an order of magnitude more restrictive as compared to current PROL limits. This would require all SSCs with potential radioactive release risks to be redesigned or would require fundamental changes to fuel design which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.4.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4.1 Internal hazards
Requirement Assessed	SSCs important to safety shall be designed and located in a manner that minimizes the probability and effects of hazards (e.g., fires and explosions) caused by external or internal events. The plant design shall take into account the potential for internal hazards, such as flooding, missile generation, pipe whip, jet impact, fire, smoke, and combustion by-products, or release of fluid from failed systems or from other installations on the site. Appropriate preventive and mitigation measures shall be provided to ensure that nuclear safety is not compromised. Internal events which the plant is designed to withstand shall be identified, and AOOs, DBAs and DECs shall be determined from these events. The possible interaction of external and internal events shall be considered, such as external events initiating internal fires or floods, or that may lead to the generation of missiles. Guidance The design should take into account specific loads and environmental conditions (temperature, pressure, humidity, radiation) imposed on structures or components by internal hazards. The following potential initiators of flooding should be considered: • leaks and breaks in pressure-retaining components • flooding by water from neighbouring buildings • spurious actuation of the fire-fighting system • overfilling of tanks • failures of isolating devices The design considers internal missiles which can be generated by failure of rotating components (such as turbines), or by failure of pressurized components. For those potential missiles considered to be credible, the following actions should be taken: • a realistic assessment is made of the postulated missile size and energy, and its potential trajectories • potentially impacted components associated with systems required to achieve and maintain a safe shutdown state are identified • a loss of these potentially impacted components is evaluated to determine if sufficient redundancy remains to achieve and maintain a safe

	shutdown state The civil design takes into account loads generated by internal hazards in the environmental loading category consistent with section 7.15.
Macro-Gap	SF01-01-15
Issue/Gap Description	Since the current design documentation does not consider internal events as leading to AOOs, DBAs and DECs, this is assessed as a gap in Clause 7.4 (Gap).
Rationale	Bruce A plant design basis, documented in the Bruce A Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. A systematic identification of internal hazards in accordance with current expectations has not been performed for the Bruce A original design. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. as those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-004 Improve fire protection provisions to achieve alignment with N293-07 (complete) GIO-019 Assess and improve seismic margins GIO-087 Bruce A Fire Protection Upgrades to Align with CSA-N293-14 (complete). Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not possible to make wholesale design changes to protect against internal events and particularly hazards for the SSCs important to safety in the current plant and specifically those associated with DECs. It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.4.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4.1 Internal hazards
Requirement Assessed	SSCs important to safety shall be designed and located in a manner that minimizes the probability and effects of hazards (e.g., fires and explosions) caused by external or internal events. The plant design shall take into account the potential for internal hazards, such as flooding, missile generation, pipe whip, jet impact, fire, smoke, and combustion by-products, or release of fluid from failed systems or from other installations on the site. Appropriate preventive and mitigation measures shall be provided to ensure that nuclear safety is not compromised. Internal events which the plant is designed to withstand shall be identified, and AOOs, DBAs and DECs shall be determined from these events. The possible interaction of external and internal events shall be considered, such as external events initiating internal fires or floods, or that may lead to the generation of missiles. Guidance The design should take into account specific loads and environmental conditions (temperature, pressure, humidity, radiation) imposed on structures or components by internal hazards. The following potential initiators of flooding should be considered: • leaks and breaks in pressure-retaining components • flooding by water from neighbouring buildings • spurious actuation of the fire-fighting system • overfilling of tanks • failures of isolating devices The design considers internal missiles which can be generated by failure of rotating components (such as turbines), or by failure of pressurized components. For those potential missiles considered to be credible, the following actions should be taken: • a realistic assessment is made of the postulated missile size and energy, and its potential trajectories • potentially impacted components associated with systems required to achieve and maintain a safe shutdown state are identified • a loss of these potentially impacted components is evaluated to determine if sufficient redundancy remains to achieve and maintain a safe

	shutdown state The civil design takes into account loads generated by internal hazards in the environmental loading category consistent with section 7.15.
Macro-Gap	SF01-01-16
Issue/Gap Description	Since the current design documentation does not consider internal events as leading to AOOs, DBAs and DECs, this is assessed as a gap in Clause 7.4 (Gap).
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. A systematic identification of internal hazards in accordance with current expectations has not been performed for the Bruce B original design. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. as those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-003 Assess pipe whip and jet impingement GIO-004 Improve fire protection provisions to achieve alignment with N293-07 (complete) GIO-019 Assess and improve seismic margin GIO-092 Bruce B Fire Protection Upgrades to Align with CSA-N293-14 Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not possible to make wholesale design changes to protect against internal events and particularly hazards for the SSCs important to safety in the current plant and specifically those associated with DECs. It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1 This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.5_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.5 Design rules and limits
Requirement Assessed	The design authority shall specify the engineering design rules for all SSCs. These rules shall comply with appropriate accepted engineering practices. The design shall also identify SSCs to which design limits are applicable. These design limits shall be specified for operational states, DBAs and DECs. Guidance Methods to ensure a robust design are applied, and proven engineering practices are adhered to in the design, as a way to ensure that the fundamental safety functions would be achieved in all operational states, DBAs and DECs. The engineering design rules for all SSCs should be determined based on their importance to safety, as determined using the criteria in section 7.1. The design rules should include, as applicable: • identified codes and standards • conservative safety margins • reliability and availability: • material selection • single-failure criterion • redundancy • separation • diversity • independence • fail-safe design • equipment qualification: • environmental qualification • seismic qualification • qualification against electromagnetic interference • operational considerations: • testability • inspectability • maintainability • aging management • management system The design of complementary design features should be such that they are effective for fulfilling the actions credited in the safety analysis, with a reasonable degree of confidence. Other SSCs that are credited for DECs should also meet this expectation.

	Design rules should include relevant national and international codes and standards. In cases of SSCs for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar SSCs may be applied; in the absence of such codes and standards, the results of experience, tests, analysis or a combination of these may be applied, and this approach should be justified. A set of design limits consistent with the key physical parameters for each SSC important to safety for the nuclear power plant should be specified for all operational states, DBAs and DECs. The design limits specified are consistent with relevant national and international codes and standards.
Macro-Gap	SF01-01-15
Issue/Gap Description	The current design documentation does not list design limits for DECs; hence this is identified as a gap (Gap).
Rationale	Bruce A plant design basis, documented in the Bruce A Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant. Design Extension Condition (DEC) is a plant state introduced as a subset of BDBAs that have to be considered in the design of a new plant. Design requirements associated with such a change affects the plant design as a whole and requires definition of a set of new design limits and acceptance criteria to be established for a specified set of SSCs as a pre-requisite. This would require a fundamentally different approach in the design and implementation of changes for the prevention and mitigation of radiation hazards associated with DECs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.5_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.5 Design rules and limits
Requirement Assessed	The design authority shall specify the engineering design rules for all SSCs. These rules shall comply with appropriate accepted engineering practices. The design shall also identify SSCs to which design limits are applicable. These design limits shall be specified for operational states, DBAs and DECs. Guidance Methods to ensure a robust design are applied, and proven engineering practices are adhered to in the design, as a way to ensure that the fundamental safety functions would be achieved in all operational states, DBAs and DECs. The engineering design rules for all SSCs should be determined based on their importance to safety, as determined using the criteria in section 7.1. The design rules should include, as applicable: • identified codes and standards • conservative safety margins • reliability and availability: • material selection • single-failure criterion • redundancy • separation • diversity • independence • fail-safe design • equipment qualification: • environmental qualification • seismic qualification • qualification against electromagnetic interference • operational considerations: • testability • inspectability • maintainability • aging management • management system The design of complementary design features should be such that they are effective for fulfilling the actions credited in the safety analysis, with a reasonable degree of confidence. Other SSCs that are credited for DECs should also meet this expectation.

	Design rules should include relevant national and international codes and standards. In cases of SSCs for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar SSCs may be applied; in the absence of such codes and standards, the results of experience, tests, analysis or a combination of these may be applied, and this approach should be justified. A set of design limits consistent with the key physical parameters for each SSC important to safety for the nuclear power plant should be specified for all operational states, DBAs and DECs. The design limits specified are consistent with relevant national and international codes and standards.
Macro-Gap	SF01-01-16
Issue/Gap Description	The current design documentation does not list design limits for DECs; hence this is identified as a gap (Gap).
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant. Design Extension Condition (DEC) is a plant state introduced as a subset of BDBAs that have to be considered in the design of a new plant. Design requirements associated with such a change affects the plant design as a whole and requires definition of a set of new design limits and acceptance criteria to be established for a specified set of SSCs as a pre-requisite. This would require a fundamentally different approach in the design and implementation of changes for the prevention and mitigation of radiation hazards associated with DECs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.6.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.2 Single-failure criterion
Requirement Assessed	All safety groups shall function in the presence of a single failure. The single-failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure as well as: 1. all failures caused by that single failure 2. all identifiable but non-detectable failures, including those in the non-tested components 3. all failures and spurious system actions that cause (or are caused by) the PIE Each safety group shall be able to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. Analysis of all possible single failures, and all associated consequential failures, shall be conducted for each component of each safety group until all safety groups have been considered. Unintended actions and failure of passive components shall be considered as two of the modes of failure of a safety group. The single failure shall be assumed to occur prior to the PIE, or at any time during the mission time for which the safety group is required to function following the PIE. Passive components may be exempt from this requirement. Exceptions to the single-failure criterion shall be infrequent, and clearly justified. Exemptions for passive components may be applied only to those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by the PIE. Design documentation shall include justification of such exemptions, by analysis, testing or a combination of analysis and testing. The justification shall take loads and environmental conditions into account as well as the total period of time after the PIE for which the functioning of the component is necessary. Check valves shall be considered to be active components if they must change state following a PIE.

Guidance

The application of the single-failure criterion (SFC) in design should follow a systematic approach applied to all safety groups. The approach should be adequately verified, such as by using failure modes and effects analysis. The SSCs inside the safety group should include both the primary SSCs and the supporting SSCs.

The detectability of failures is implicit in the application of the SFC. Detectability is a function of the system design and the specified tests. A failure that cannot be detected through periodic testing, or revealed by alarm or anomalous indication, is non-detectable. An objective in a single-failure analysis is to identify non-detectable failures. To deal with identifiable but non-detectable failures, the following actions should be considered:

- preferred action: the system or the test scheme should be redesigned to make the failure detectable
- alternative action: when analyzing the effect of each single failure, all identified non- detectable failures should be assumed to have occurred. Therefore, the design should take appropriate measures to address these non-detectable failures, such as adequate redundancy and diversity

Justification in support of an exception to the SFC should consider the consequences of failure, practicality of alternatives, added complexity and operational considerations. The integrated effect of all exceptions should not significantly degrade safety; in particular, defence in depth should be preserved.

For passive components that are exempt from the SFC, the following should be considered in order to demonstrate a high degree of performance assurance:

- adequate testing during the manufacturing stage
- sample testing from those components received from the manufacturer
- adequate testing during construction and commissioning stages
- necessary testing to verify their reliability after the components have been removed from service during the operation stage

Any consideration for an exception to the SFC during testing and maintenance should fall into one of the following permissible categories:

- the safety function is provided by two redundant, independent systems (e.g., two redundant, fully effective, independent cooling means)
- the expected duration of testing and maintenance is shorter than the time available before the function is required following an initiating event (e.g., spent fuel storage pool cooling)
- the loss of safety function is partial and unlikely to lead to

	significant increase in risk even in the event of failure (e.g., small area containment isolation) • the loss of system redundancy has minor safety significance (e.g., control room air filtering) • the loss of system redundancy may slightly increase PIE frequency, but does not impact accident progression (e.g., leak detection) A request for an exception during testing and maintenance should also be supported by a satisfactory reliability argument covering the allowable outage time. The OLCs should clearly state the allowable testing and maintenance time, along with any additional operational restrictions, such as suspension of additional testing or maintenance on a backup system for the duration of the exception.
Macro-Gap	SF01-05-15
Issue/Gap Description	A review of the same clause in RD-337 indicated that the Bruce A design does not fully meet this requirement, as documented in [NK21-CORR-00531-11005].The application of the single failure criterion for the Bruce A design reflects the interpretation of this criterion that was prevalent at that time, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. This does not follow the newer, more restrictive, interpretations of the single failure criterion; therefore is assessed as a gap (Gap).
Rationale	Bruce Power is in compliance with the current design basis which reflects the interpretation of single failure criterion that has been accepted in the current licensing basis, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. Application of the newer and more restrictive interpretations of the single failure criterion would lead to fundamental design changes both at the system and component level which is impracticable. Note that assessment of single failures is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.6.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.2 Single-failure criterion
Requirement Assessed	All safety groups shall function in the presence of a single failure. The single-failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure as well as: 1. all failures caused by that single failure 2. all identifiable but non-detectable failures, including those in the non-tested components 3. all failures and spurious system actions that cause (or are caused by) the PIE Each safety group shall be able to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. Analysis of all possible single failures, and all associated consequential failures, shall be conducted for each component of each safety group until all safety groups have been considered. Unintended actions and failure of passive components shall be considered as two of the modes of failure of a safety group. The single failure shall be assumed to occur prior to the PIE, or at any time during the mission time for which the safety group is required to function following the PIE. Passive components may be exempt from this requirement. Exceptions to the single-failure criterion shall be infrequent, and clearly justified. Exemptions for passive components may be applied only to those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by the PIE. Design documentation shall include justification of such exemptions, by analysis, testing or a combination of analysis and testing. The justification shall take loads and environmental conditions into account as well as the total period of time after the PIE for which the functioning of the component is necessary. Check valves shall be considered to be active components if they must change state following a PIE.

	Guidance The application of the single-failure criterion (SFC) in design should follow a systematic approach applied to all safety groups. The approach should be adequately verified, such as by using failure modes and effects analysis. The SSCs inside the safety group should include both the primary SSCs and the supporting SSCs. The detectability of failures is implicit in the application of the SFC. Detectability is a function of the system design and the specified tests. A failure that cannot be detected through periodic testing, or revealed by alarm or anomalous indication, is non-detectable. An objective in a single-failure analysis is to identify non-detectable failures. To deal with identifiable but non-detectable failures, the following actions should be considered: • preferred action: the system or the test scheme should be redesigned to make the failure detectable • alternative action: when analyzing the effect of each single failure, all identified non- detectable failures should be assumed to have occurred. Therefore, the design should take appropriate measures to address these non-detectable failures, such as adequate redundancy and diversity Justification in support of an exception to the SFC should consider the consequences of failure, practicality of alternatives, added complexity and operational considerations. The integrated effect of all exceptions should not significantly degrade safety; in particular, defence in depth should be preserved. For passive components that are exempt from the SFC, the following should be considered in order to demonstrate a high degree of performance assurance: • adequate testing during the manufacturing stage • sample testing from those components received from the manufacturer • adequate testing during construction and commissioning stages • necessary testing to verify their reliability after the components have been removed from service during the operation stage Any consideration for an exception to the SFC during testing and maintenance should fall into one of the following permissible categories: • the safety function is provided by two redundant, independent systems (e.g., two redundant, fully effective, independent cooling means) • the expected duration of testing and maintenance is shorter than the time available before the function is required following an initiating event (e.g., spent fuel storage pool cooling) • the loss of safety function is partial and unlikely to lead to

	significant increase in risk even in the event of failure (e.g., small area containment isolation) • the loss of system redundancy has minor safety significance (e.g., control room air filtering) • the loss of system redundancy may slightly increase PIE frequency, but does not impact accident progression (e.g., leak detection) A request for an exception during testing and maintenance should also be supported by a satisfactory reliability argument covering the allowable outage time. The OLCs should clearly state the allowable testing and maintenance time, along with any additional operational restrictions, such as suspension of additional testing or maintenance on a backup system for the duration of the exception.
Macro-Gap	SF01-05-16
Issue/Gap Description	A review of the same clause in RD-337 indicated that the Bruce A and B design does not fully meet this requirement, as documented in [NK21-CORR-00531-11005 / NK29-CORR-00531-11397].The application of the single failure criterion for the Bruce A and B design reflects the interpretation of this criterion that was prevalent at that time, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. This does not follow the newer, more restrictive, interpretations of the single failure criterion; therefore is assessed as a gap (Gap 1).
Rationale	Bruce Power is in compliance with the current design basis which reflects the interpretation of single failure criterion that has been accepted in the current licensing basis, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. Application of the newer and more restrictive interpretations of the single failure criterion would lead to fundamental design changes both at the system and component level which is impracticable. Note that assessment of single failures is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.6.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.2 Single-failure criterion
Requirement Assessed	All safety groups shall function in the presence of a single failure. The single-failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure as well as: 1. all failures caused by that single failure 2. all identifiable but non-detectable failures, including those in the non-tested components 3. all failures and spurious system actions that cause (or are caused by) the PIE Each safety group shall be able to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. Analysis of all possible single failures, and all associated consequential failures, shall be conducted for each component of each safety group until all safety groups have been considered. Unintended actions and failure of passive components shall be considered as two of the modes of failure of a safety group. The single failure shall be assumed to occur prior to the PIE, or at any time during the mission time for which the safety group is required to function following the PIE. Passive components may be exempt from this requirement. Exceptions to the single-failure criterion shall be infrequent, and clearly justified. Exemptions for passive components may be applied only to those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by the PIE. Design documentation shall include justification of such exemptions, by analysis, testing or a combination of analysis and testing. The justification shall take loads and environmental conditions into account as well as the total period of time after the PIE for which the functioning of the component is necessary. Check valves shall be considered to be active components if they must change state following a PIE.

	Guidance The application of the single-failure criterion (SFC) in design should follow a systematic approach applied to all safety groups. The approach should be adequately verified, such as by using failure modes and effects analysis. The SSCs inside the safety group should include both the primary SSCs and the supporting SSCs. The detectability of failures is implicit in the application of the SFC. Detectability is a function of the system design and the specified tests. A failure that cannot be detected through periodic testing, or revealed by alarm or anomalous indication, is non-detectable. An objective in a single-failure analysis is to identify non-detectable failures. To deal with identifiable but non-detectable failures, the following actions should be considered: • preferred action: the system or the test scheme should be redesigned to make the failure detectable • alternative action: when analyzing the effect of each single failure, all identified non- detectable failures should be assumed to have occurred. Therefore, the design should take appropriate measures to address these non-detectable failures, such as adequate redundancy and diversity Justification in support of an exception to the SFC should consider the consequences of failure, practicality of alternatives, added complexity and operational considerations. The integrated effect of all exceptions should not significantly degrade safety; in particular, defence in depth should be preserved. For passive components that are exempt from the SFC, the following should be considered in order to demonstrate a high degree of performance assurance: • adequate testing during the manufacturing stage • sample testing from those components received from the manufacturer • adequate testing during construction and commissioning stages • necessary testing to verify their reliability after the components have been removed from service during the operation stage Any consideration for an exception to the SFC during testing and maintenance should fall into one of the following permissible categories: • the safety function is provided by two redundant, independent systems (e.g., two redundant, fully effective, independent cooling means) • the expected duration of testing and maintenance is shorter than the time available before the function is required following an initiating event (e.g., spent fuel storage pool cooling) • the loss of safety function is partial and unlikely to lead to

	significant increase in risk even in the event of failure (e.g., small area containment isolation) • the loss of system redundancy has minor safety significance (e.g., control room air filtering) • the loss of system redundancy may slightly increase PIE frequency, but does not impact accident progression (e.g., leak detection) A request for an exception during testing and maintenance should also be supported by a satisfactory reliability argument covering the allowable outage time. The OLCs should clearly state the allowable testing and maintenance time, along with any additional operational restrictions, such as suspension of additional testing or maintenance on a backup system for the duration of the exception.
Macro-Gap	SF01-03-16
Issue/Gap Description	For Bruce B design there is no systematic analysis of all possible single failures, and all associated consequential failures, conducted for each component of each safety group as required in this clause. Therefore, this is assessed as a gap (Gap 2).
Rationale	The application of the single failure criterion for the Bruce A and B design reflects the interpretation of this criterion that was prevalent at that time, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. Design of each safety group to perform all safety functions required for a PIE in the presence of any single component failure with the application of the newer and more restrictive interpretations of the single failure criterion would lead to fundamental design changes both at the system and component level which is impracticable. Note that assessment of single failures is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_7.6.5.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.5.2 Sharing of SSCs between reactors
Requirement Assessed	SSCs important to safety shall typically not be shared between two or more reactors. In exceptional cases when SSCs are shared between two or more reactors, such sharing shall exclude safety systems and turbine generator buildings that contain high-pressure steam and feedwater systems, unless this contributes to enhanced safety. If sharing of SSCs between reactors is arranged, then the following requirements shall apply: 1. safety requirements shall be met for all reactors during operational states, DBAs and DECs 2. in the event of an accident involving one of the reactors, orderly shutdown, cool down, and removal of residual heat shall be achievable for the other reactor(s) When an NPP is under construction adjacent to an operating plant, and the sharing of SSCs between reactors has been justified, the availability of the SSCs and their capacity to meet all safety requirements for the operating units shall be assessed during the construction phase.
Macro-Gap	SF01-20-16
Issue/Gap Description	Bruce B design includes sharing of special safety systems between reactors without justification that such sharing contributed to enhanced safety as required in this clause. Therefore it is assessed as a design gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis which includes shared safety systems as documented in the Bruce B Safety Report. The early design philosophy used for the multi-unit stations in Canada was to share some of the systems that were important to safety. The sharing of systems was factored into the reliability requirements of these systems and each has redundant components to ensure adequate reliability. The accident analyses and the PRA recognize the shared functions and have shown that the design is adequate to meet Bruce Power's safety goals and the requirements of the PROL. Exclusion of SSCs important to safety between one or more reactors as well as exclusion of sharing safety systems and turbine generator buildings that contain high-pressure steam and feedwater systems would lead to fundamental design changes both at the plant layout, system and component level which is impracticable.

	Justification of sharing is demonstrated by safety analysis which is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.1 Reactor core
Requirement Assessed	Reactor core parameters and their limits shall be specified. The design shall consider all foreseeable reactor core configurations for normal operation. The reactor core, including the fuel elements, reactivity control mechanisms, reflectors, fuel channel and structural parts, shall be designed so that the reactor can be shutdown, cooled and held subcritical with an adequate margin in operational states, DBAs and DECs. The anticipated upper limit of possible deformation or other changes due to irradiation conditions shall be evaluated. These evaluations shall be supported by data from experiments, and from experience with irradiation. The design shall provide protection against those deformations, or any other changes to reactor structures that have the potential to adversely affect the behaviour of the core or associated systems. The reactor core and associated structures and cooling systems shall: 1. withstand static and dynamic loading, including thermal expansion and contraction 2. withstand vibration (such as flow-induced and acoustic vibration) 3. ensure chemical compatibility, including service-related contaminants 4. meet thermal material limits 5. meet radiation damage limits The reactor core design shall include provisions for a guaranteed shutdown state as described in section 7.11. The design of the core shall be such that: 1. the fission chain reaction is controlled during operational states 2. the maximum degree of positive reactivity and its maximum rate of increase by insertion in operational states and DBAs are limited by a combination of the inherent neutronic characteristics of the core, its thermal-hydraulic characteristics, and the capabilities of the control system and means of shutdown, so that no resultant failure of the reactor pressure boundary will occur, cooling capability will be maintained, and no significant damage will occur to the reactor core

	The shutdown margin for all shutdown states shall be such that the core will remain subcritical for any credible changes in the core configuration and reactivity addition. If operator intervention is required to keep the reactor in a shutdown state, the feasibility, timeliness, and effectiveness of such intervention shall be demonstrated. Guidance on nuclear design The design of the reactor core should provide confidence that the permissible design limits, under operational states, DBAs and DECs, are not exceeded, taking into account engineering tolerances and uncertainties associated with the calculations. The nuclear design deals with flux and power distribution within the reactor core, the design and use of reactivity control systems for normal operation and for shutting down the reactor, core stability, the various reactivity feedback characteristics, and the physics of the fuel. The design of the reactor core and associated coolant and fuel systems should take into account all practical means so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity and power. The consequences of those accidents that would be aggravated by a positive reactivity feedback should be either acceptable, or be satisfactorily mitigated by other design features. The design should take into account measurements made in previous reactors and critical experiments and their use in the uncertainty analyses. The design should define the measurements to be made, including start-up confirmatory tests and periodically required measurements. The design should provide for I&C to: • maintain the variables and systems within prescribed operating ranges • monitor variables and systems that can affect the fission process over anticipated ranges for operational states, DBAs and DECs These I&Cs should be demonstrated to be effective. Defence in depth The nuclear design should incorporate inherently safe features to reduce the reliance on engineered safety systems or operational procedures. Defence in depth and related principles should be applied in the design of the reactivity control safety function, such that the fission chain reaction is controlled during operational states, and, when necessary, terminated for DBAs and DECs.

	The nuclear design should provide for effective means to ensure success of the following safety functions to: • prevention of unacceptable reactivity transients • shutdown of the reactor as necessary to prevent progression of AOOs to DBAs, or DBAs to DECs • maintain and monitor the reactor in a safe shutdown state Core power densities and distributions The design limits for the power densities and power distributions should be determined from an integrated consideration of fuel design limits, thermal limits, decay heat limits, and AOO and accident analyses. For power distribution, the reactor core design should demonstrate the following: • There is a high level of confidence that the proposed design limits can be met within the expected operational range of the reactor, taking into account: • the analytical methods and data for the design calculations • uncertainty analyses and experimental comparisons presented for the design calculations • the sufficiency of design cases calculated covering times in fuel reload cycle, or during on-power fuelling (depending upon the reactor design, reactivity devices configurations, and load-follow transients) • special problems (such as power spikes due to densification), possible asymmetries, and misaligned reactivity devices • There is a high level of confidence that, during normal operation, the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation. The processing of that information should include: • calculations (instrument-calculation correlations) involved in the processing • operating procedures used • the requirements for periodic check measurements • the accuracy of design calculations used in developing correlations when primary variables are not directly measured • the uncertainty analyses for the information and processing system • the requirements for instruments, the calibration and calculations involved in their use, and the uncertainties involved in conversion of instrument readings into power distribution • the limits and set points for control actions, alarms, or automatic trip for instrument systems and demonstration that these systems can maintain the reactor within design power distribution limits (including the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., flux tilt alarms) • measurements in previous reactors and critical experiments, including their use in the uncertainty analyses

	• measurements needed for start-up confirmatory tests and the required periodical measurements The limiting power distributions should be determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic shutdown), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed. The design should establish the correlation between design power distributions and operating power distributions, including instrument-calculation correlations, operating procedures used, and measurements that will be taken. Necessary limits on these operations should be established. The breakdown of design power distributions into the following components should be established: • power generated in the fuel • power generated directly in the coolant and moderator • power generated directly in the core internals The reference design core power distributions (axial, radial, and local distributions and peaking factors) used in AOO and accident analyses should be established. In addition, power distributions within fuel pins should be established. The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during AOOs and that other limits are not exceeded during DBAs and DECs. The design limits, along with related uncertainties, operating limits, instrument requirements, and set-points, should be incorporated into OLCs. Reactivity coefficients The design should establish and characterize the bounding reference values for reactivity coefficients. These reference values should be conservative. The range of plant states to be covered should include the entire operating range – from cold shutdown through full power – and the extremes reached in AOOs, DBAs and DECs. It should include the full range of the fuelling cycle, and an appropriate range of reactivity device configurations. The design calculations of reactivity coefficients should cover the full applicable range of the variables and modelling approximations in AOO and accident analyses, including approximations related to modelling and

	nodalization of the reactor cooling system. Where applicable, the difference between intra- and inter-assembly moderator coefficients needs to be established. Conservatism should be considered based on: • the use of a coefficient (i.e., the analyses in which it is important) • whether state of the art tools have been used for calculation of the coefficient • the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors • any required checks of the coefficient in the start-up program following significant core reconfiguration The design calculation should cover and be supported by the following: • calculated nominal values for the reactivity coefficients, such as the coolant and moderator coefficients (temperature, void, or density coefficients), the Doppler coefficient and power coefficients • uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude (by examination of the accuracy of the methods used in calculations), and comparison, where possible, with reactor experiments. • combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady-state analysis (primarily control requirements), stability analyses, and the AOO and accident analyses For comparisons to experiments, it is important to show that the experiments are applicable and relevant, and the experimental conditions overlap the operating and anticipated accident conditions. It is recognized that reactivity coefficients of the design are important in determining the reactor behavior and safety characteristics. This document does not have specific requirements on the sign or magnitude of the reactivity coefficients including the power coefficient of reactivity. Instead, this document requires a number of design provisions related to the nuclear design to ensure that the design is acceptable for reactor control, stability and plant safety. If a reactor design has a positive power coefficient of reactivity for any operating state, the design authority should demonstrate that operation with a positive power coefficient is acceptable, by showing: • a bounding value of power coefficient of reactivity has been calculated for all permitted operating states and used in control, stability, and safety analyses • measurements of the power coefficient of reactivity are conducted at start-up and periodically for certain operating limiting core conditions to demonstrate that measured values are bounded by calculated values with adequate margin

	• the reactor control system is designed with adequate reliability and has the capability to automatically accommodate for a positive power coefficient of reactivity for a wide range of AOOs The design should ensure that the likelihood of exceeding specified criteria of the AOOs without shutdown is sufficiently small, by demonstrating either that the criteria are met, or that a diverse shutdown means is installed, which reduces significantly the probability of a failure to shutdown. Criticality The nuclear design should ensure that the criticality of the reactor during refuelling is controlled. If on-power refuelling is used to compensate for core reactivity depletion, the nuclear design should establish the values of core excess reactivity, maximum local powers, amount of fuel loaded per refuelling operation and frequency of refuelling load. The design should also ensure that the maximum core excess reactivity and predicted local power peaks will not exceed the control system capability and fuel thermal limits. Core stability Power oscillations that could result in conditions exceeding specified acceptable fuel design limits should be reliably and readily detected and suppressed. Assessment of reactor core stability should include: • phenomena and reactor aspects that influence the stability of the nuclear reactor core • calculations and considerations given to xenon-induced spatial oscillations • potential stability issues, due to other phenomena or conditions • verification of the analytical methods for comparison with measured data Analytical methods The analytical methods and database used for nuclear design and reactor physics analyses should be consistent with modern best practices. Also, the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and ranges of key parameters in the validation database should overlap those expected in design and safety analysis. The design should be such that the analytical methods used in the nuclear design (including those for predicting criticality, reactivity coefficients, burnup and stability) as well as the database and nuclear data libraries used for neutron cross-section data and other nuclear parameters (including delayed neutron and photo neutron data and other relevant

	data) are adequate and fit for application, based on adequate qualification. The qualification should be based on proven practices for validation and verification, using the acceptable codes and standards. A validation or verification method can be proven either by meeting accepted verification and validation standards, or by established practice, or some combination of these. New method(s) are “proven” by performing a number of acceptance and demonstration tests that show the method(s) meets pre-defined criteria. Core internals and vessel The nuclear design should establish: • neutron flux spectrum above 1 million electron volts (MeV) in the core, at the core boundaries, and at the inside vessel wall, if applicable • assumptions used in the calculations, these include the power level, the use factor, the type of fuel cycle considered, and the design life of the vessel • computer codes used in the analysis • the database for fast neutron cross-sections • the geometric modelling of the reactor core, internals, and vessel(s) • uncertainties in the calculations Guidance on core management and fuel handling The reactor design should be such that the plant will operate within the specified operating limits for the entire reactor lifecycle (including intermediate reactor core states). The design should provide for functional tests to be performed periodically for monitoring the health of the reactor components. The design should provide for the capability to monitor online important core parameters, to ensure that the acceptable operating limits for the reactor are not exceeded during normal operation. The types of detectors and other devices used in monitoring the core parameters should be described. The reactor control strategy should be defined, to ensure that the reactor will be restored to an acceptable safe state if any reactor parameter deviates from its allowed domain. The control strategy should be such that fuel integrity will be maintained for all AOOs. The refuelling scheme should be developed to ensure that the intermediate refuelling configurations do not have more reactivity than the most reactive configuration approved in the design. The core parameters

	for the intermediate configurations should be within their approved limits. The design should allow for data acquisition during reactor operation and record-keeping for later retrieval and analysis. The design should take into account the details of fuel management strategy including the loading of fuel into the fresh core, and the criteria for determining the location of fuel assemblies to be unloaded from the reactor and loaded with fresh fuel. For reactor designs where a significant fraction of the fuel is replaced or shuffled during fuelling, the design should provide for diagnostic tests at startup. These tests should verify that the core parameters are within their allowed range. Guidance on mechanical design of reactor internals The reactor internals classified as core support structures according to the ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, NG-1121, Core Support Structures, should be designed, fabricated, and examined in accordance with the provisions of ASME BPVC Section III Division 1, subsection NG. Those reactor internals not classified as ASME BPVC Code, Section III, Division 1, Core Support Structures should be classified as internal structures in accordance with ASME Code, Section III, Division 1, Subsection NG-1122. The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals (other than the core support structures) should meet the guidelines of ASME Code, Section III, Division 1, Subsection NG-3000, and be constructed so as to not adversely affect the integrity of the core support structures. If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified in the design. For non-ASME code structures, components and supports, design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. Any decreases in design margins should be justified. Specific reactor internals of a high safety class should be designed, fabricated, and examined in accordance with the applicable codes and standards, such as ASME Section III for light water reactors (LWR), and CSA N285.0, General Requirements for Pressure-retaining Systems and Components in CANDU Nuclear Power Plants for CANDU.
Macro-Gap	SF01-01-15
Issue/Gap Description	The design limits and margins as required in the second paragraph of this

	clause (i.e., reactor core, including the fuel elements, reactivity control mechanisms, etc.) for DECs cannot be confirmed in the existing design documentation. Therefore, it is assessed as a gap. (Gap).
Rationale	Bruce A plant design basis, documented in the Bruce A Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as DECs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant It is impracticable to make design changes to the reactor core, including the fuel elements, reactivity control mechanisms, reflectors, fuel channel and structural to ensure that the reactor can be shut down, cooled and held subcritical with an adequate margin during DECs which have not been explicitly defined and considered as part of the original design. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.1 Reactor core
Requirement Assessed	Reactor core parameters and their limits shall be specified. The design shall consider all foreseeable reactor core configurations for normal operation. The reactor core, including the fuel elements, reactivity control mechanisms, reflectors, fuel channel and structural parts, shall be designed so that the reactor can be shutdown, cooled and held subcritical with an adequate margin in operational states, DBAs and DECs. The anticipated upper limit of possible deformation or other changes due to irradiation conditions shall be evaluated. These evaluations shall be supported by data from experiments, and from experience with irradiation. The design shall provide protection against those deformations, or any other changes to reactor structures that have the potential to adversely affect the behaviour of the core or associated systems. The reactor core and associated structures and cooling systems shall: 1. withstand static and dynamic loading, including thermal expansion and contraction 2. withstand vibration (such as flow-induced and acoustic vibration) 3. ensure chemical compatibility, including service-related contaminants 4. meet thermal material limits 5. meet radiation damage limits The reactor core design shall include provisions for a guaranteed shutdown state as described in section 7.11. The design of the core shall be such that: 1. the fission chain reaction is controlled during operational states 2. the maximum degree of positive reactivity and its maximum rate of increase by insertion in operational states and DBAs are limited by a combination of the inherent neutronic characteristics of the core, its thermal-hydraulic characteristics, and the capabilities of the control system and means of shutdown, so that no resultant failure of the reactor pressure boundary will occur, cooling capability will be maintained, and no significant damage will occur to the reactor core

	The shutdown margin for all shutdown states shall be such that the core will remain subcritical for any credible changes in the core configuration and reactivity addition. If operator intervention is required to keep the reactor in a shutdown state, the feasibility, timeliness, and effectiveness of such intervention shall be demonstrated. Guidance on nuclear design The design of the reactor core should provide confidence that the permissible design limits, under operational states, DBAs and DECs, are not exceeded, taking into account engineering tolerances and uncertainties associated with the calculations. The nuclear design deals with flux and power distribution within the reactor core, the design and use of reactivity control systems for normal operation and for shutting down the reactor, core stability, the various reactivity feedback characteristics, and the physics of the fuel. The design of the reactor core and associated coolant and fuel systems should take into account all practical means so that, in the power operating range, the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity and power. The consequences of those accidents that would be aggravated by a positive reactivity feedback should be either acceptable, or be satisfactorily mitigated by other design features. The design should take into account measurements made in previous reactors and critical experiments and their use in the uncertainty analyses. The design should define the measurements to be made, including start-up confirmatory tests and periodically required measurements. The design should provide for I&C to: • maintain the variables and systems within prescribed operating ranges • monitor variables and systems that can affect the fission process over anticipated ranges for operational states, DBAs and DECs These I&Cs should be demonstrated to be effective. Defence in depth The nuclear design should incorporate inherently safe features to reduce the reliance on engineered safety systems or operational procedures. Defence in depth and related principles should be applied in the design of the reactivity control safety function, such that the fission chain reaction is controlled during operational states, and, when necessary, terminated for DBAs and DECs.

	The nuclear design should provide for effective means to ensure success of the following safety functions to: • prevention of unacceptable reactivity transients • shutdown of the reactor as necessary to prevent progression of AOOs to DBAs, or DBAs to DECs • maintain and monitor the reactor in a safe shutdown state Core power densities and distributions The design limits for the power densities and power distributions should be determined from an integrated consideration of fuel design limits, thermal limits, decay heat limits, and AOO and accident analyses. For power distribution, the reactor core design should demonstrate the following: • There is a high level of confidence that the proposed design limits can be met within the expected operational range of the reactor, taking into account: • the analytical methods and data for the design calculations • uncertainty analyses and experimental comparisons presented for the design calculations • the sufficiency of design cases calculated covering times in fuel reload cycle, or during on-power fuelling (depending upon the reactor design, reactivity devices configurations, and load-follow transients) • special problems (such as power spikes due to densification), possible asymmetries, and misaligned reactivity devices • There is a high level of confidence that, during normal operation, the design limits will not be exceeded, based on consideration of information received from the power distribution monitoring instrumentation. The processing of that information should include: • calculations (instrument-calculation correlations) involved in the processing • operating procedures used • the requirements for periodic check measurements • the accuracy of design calculations used in developing correlations when primary variables are not directly measured • the uncertainty analyses for the information and processing system • the requirements for instruments, the calibration and calculations involved in their use, and the uncertainties involved in conversion of instrument readings into power distribution • the limits and set points for control actions, alarms, or automatic trip for instrument systems and demonstration that these systems can maintain the reactor within design power distribution limits (including the instrumentation alarms for the limits of normal operation (e.g., offset limits, control bank limits) and for abnormal situations (e.g., flux tilt alarms) • measurements in previous reactors and critical experiments, including their use in the uncertainty analyses

	• measurements needed for start-up confirmatory tests and the required periodical measurements The limiting power distributions should be determined such that the limits on power densities and peaking factors can be maintained in operation. These limiting power distributions may be maintained (i.e., not exceeded) administratively (i.e., not by automatic shutdown), provided a suitable demonstration is made that sufficient, properly translated information and alarms are available from the reactor instrumentation to keep the operator informed. The design should establish the correlation between design power distributions and operating power distributions, including instrument-calculation correlations, operating procedures used, and measurements that will be taken. Necessary limits on these operations should be established. The breakdown of design power distributions into the following components should be established: • power generated in the fuel • power generated directly in the coolant and moderator • power generated directly in the core internals The reference design core power distributions (axial, radial, and local distributions and peaking factors) used in AOO and accident analyses should be established. In addition, power distributions within fuel pins should be established. The design limits for power densities (and thus for peaking factors) during normal operation should be such that acceptable fuel design limits are not exceeded during AOOs and that other limits are not exceeded during DBAs and DECs. The design limits, along with related uncertainties, operating limits, instrument requirements, and set-points, should be incorporated into OLCs. Reactivity coefficients The design should establish and characterize the bounding reference values for reactivity coefficients. These reference values should be conservative. The range of plant states to be covered should include the entire operating range – from cold shutdown through full power – and the extremes reached in AOOs, DBAs and DECs. It should include the full range of the fuelling cycle, and an appropriate range of reactivity device configurations. The design calculations of reactivity coefficients should cover the full applicable range of the variables and modelling approximations in AOO and accident analyses, including approximations related to modelling and

	nodalization of the reactor cooling system. Where applicable, the difference between intra- and inter-assembly moderator coefficients needs to be established. Conservatism should be considered based on: • the use of a coefficient (i.e., the analyses in which it is important) • whether state of the art tools have been used for calculation of the coefficient • the uncertainty associated with such calculations, experimental checks of the coefficient in operating reactors • any required checks of the coefficient in the start-up program following significant core reconfiguration The design calculation should cover and be supported by the following: • calculated nominal values for the reactivity coefficients, such as the coolant and moderator coefficients (temperature, void, or density coefficients), the Doppler coefficient and power coefficients • uncertainty analyses for nominal values, including the magnitude of the uncertainty and the justification of the magnitude (by examination of the accuracy of the methods used in calculations), and comparison, where possible, with reactor experiments. • combination of nominal values and uncertainties to provide suitably conservative values for use in reactor steady-state analysis (primarily control requirements), stability analyses, and the AOO and accident analyses For comparisons to experiments, it is important to show that the experiments are applicable and relevant, and the experimental conditions overlap the operating and anticipated accident conditions. It is recognized that reactivity coefficients of the design are important in determining the reactor behavior and safety characteristics. This document does not have specific requirements on the sign or magnitude of the reactivity coefficients including the power coefficient of reactivity. Instead, this document requires a number of design provisions related to the nuclear design to ensure that the design is acceptable for reactor control, stability and plant safety. If a reactor design has a positive power coefficient of reactivity for any operating state, the design authority should demonstrate that operation with a positive power coefficient is acceptable, by showing: • a bounding value of power coefficient of reactivity has been calculated for all permitted operating states and used in control, stability, and safety analyses • measurements of the power coefficient of reactivity are conducted at start-up and periodically for certain operating limiting core conditions to demonstrate that measured values are bounded by calculated values with adequate margin

	• the reactor control system is designed with adequate reliability and has the capability to automatically accommodate for a positive power coefficient of reactivity for a wide range of AOOs The design should ensure that the likelihood of exceeding specified criteria of the AOOs without shutdown is sufficiently small, by demonstrating either that the criteria are met, or that a diverse shutdown means is installed, which reduces significantly the probability of a failure to shutdown. Criticality The nuclear design should ensure that the criticality of the reactor during refuelling is controlled. If on-power refuelling is used to compensate for core reactivity depletion, the nuclear design should establish the values of core excess reactivity, maximum local powers, amount of fuel loaded per refuelling operation and frequency of refuelling load. The design should also ensure that the maximum core excess reactivity and predicted local power peaks will not exceed the control system capability and fuel thermal limits. Core stability Power oscillations that could result in conditions exceeding specified acceptable fuel design limits should be reliably and readily detected and suppressed. Assessment of reactor core stability should include: • phenomena and reactor aspects that influence the stability of the nuclear reactor core • calculations and considerations given to xenon-induced spatial oscillations • potential stability issues, due to other phenomena or conditions • verification of the analytical methods for comparison with measured data Analytical methods The analytical methods and database used for nuclear design and reactor physics analyses should be consistent with modern best practices. Also, the experiments used to validate the analytical methods should be adequate representations of fuel designs in the reactor and ranges of key parameters in the validation database should overlap those expected in design and safety analysis. The design should be such that the analytical methods used in the nuclear design (including those for predicting criticality, reactivity coefficients, burnup and stability) as well as the database and nuclear data libraries used for neutron cross-section data and other nuclear parameters (including delayed neutron and photo neutron data and other relevant

	data) are adequate and fit for application, based on adequate qualification. The qualification should be based on proven practices for validation and verification, using the acceptable codes and standards. A validation or verification method can be proven either by meeting accepted verification and validation standards, or by established practice, or some combination of these. New method(s) are "proven" by performing a number of acceptance and demonstration tests that show the method(s) meets pre-defined criteria. Core internals and vessel The nuclear design should establish: • neutron flux spectrum above 1 million electron volts (MeV) in the core, at the core boundaries, and at the inside vessel wall, if applicable • assumptions used in the calculations, these include the power level, the use factor, the type of fuel cycle considered, and the design life of the vessel • computer codes used in the analysis • the database for fast neutron cross-sections • the geometric modelling of the reactor core, internals, and vessel(s) • uncertainties in the calculations Guidance on core management and fuel handling The reactor design should be such that the plant will operate within the specified operating limits for the entire reactor lifecycle (including intermediate reactor core states). The design should provide for functional tests to be performed periodically for monitoring the health of the reactor components. The design should provide for the capability to monitor online important core parameters, to ensure that the acceptable operating limits for the reactor are not exceeded during normal operation. The types of detectors and other devices used in monitoring the core parameters should be described. The reactor control strategy should be defined, to ensure that the reactor will be restored to an acceptable safe state if any reactor parameter deviates from its allowed domain. The control strategy should be such that fuel integrity will be maintained for all AOOs. The refuelling scheme should be developed to ensure that the intermediate refuelling configurations do not have more reactivity than the most reactive configuration approved in the design. The core parameters

	for the intermediate configurations should be within their approved limits. The design should allow for data acquisition during reactor operation and record-keeping for later retrieval and analysis. The design should take into account the details of fuel management strategy including the loading of fuel into the fresh core, and the criteria for determining the location of fuel assemblies to be unloaded from the reactor and loaded with fresh fuel. For reactor designs where a significant fraction of the fuel is replaced or shuffled during fuelling, the design should provide for diagnostic tests at startup. These tests should verify that the core parameters are within their allowed range. Guidance on mechanical design of reactor internals The reactor internals classified as core support structures according to the ASME Boiler and Pressure Vessel Code (BPVC), Section III, Division 1, NG-1121, Core Support Structures, should be designed, fabricated, and examined in accordance with the provisions of ASME BPVC Section III Division 1, subsection NG. Those reactor internals not classified as ASME BPVC Code, Section III, Division 1, Core Support Structures should be classified as internal structures in accordance with ASME Code, Section III, Division 1, Subsection NG-1122. The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals (other than the core support structures) should meet the guidelines of ASME Code, Section III, Division 1, Subsection NG-3000, and be constructed so as to not adversely affect the integrity of the core support structures. If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified in the design. For non-ASME code structures, components and supports, design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. Any decreases in design margins should be justified. Specific reactor internals of a high safety class should be designed, fabricated, and examined in accordance with the applicable codes and standards, such as ASME Section III for light water reactors (LWR), and CSA N285.0, General Requirements for Pressure-retaining Systems and Components in CANDU Nuclear Power Plants for CANDU.
Macro-Gap	SF01-01-16
Issue/Gap Description	The design limits and margins as required in the second paragraph of this

	clause (i.e., reactor core, including the fuel elements, reactivity control mechanisms, etc.) for DECs cannot be confirmed in the existing design documentation. Therefore, it is assessed as a gap. (Gap).
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as DECs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant It is impracticable to make design changes to the reactor core, including the fuel elements, reactivity control mechanisms, reflectors, fuel channel and structural to ensure that the reactor can be shut down, cooled and held subcritical with an adequate margin during DECs which have not been explicitly defined and considered as part of the original design. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.10.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.10.3 Emergency support facilities
Requirement Assessed	The design shall provide for onsite emergency support facilities that are separate from the plant control rooms for use by the technical support staff and emergency support staff in the event of an emergency. The emergency support facilities shall consist of a technical support centre (TSC) and an onsite emergency response facility. The technical support centre and the emergency response facility can be located in one place or separated. The emergency support facilities shall provide equipment, facilities, and communication means for trained staff to manage, control and coordinate any emergency response as well as to provide technical support to operations, emergency response organizations, and severe accident management evaluation. The emergency support facilities design shall ensure that appropriate lighting levels and thermal environment are maintained, and that noise levels are minimized in accordance with applicable standards and codes. The emergency support facilities shall include secure means of communication with the MCR, SCR, and other important points in the plant, and with onsite and offsite emergency response organizations. The design shall ensure that the emergency support facilities: 1. includes provisions to protect occupants over protracted periods from the hazards resulting from DBAs and DECs 2. is equipped with adequate facilities to allow extended operating periods The emergency response facility shall include a SPDS similar to those in the MCR and in the SCR. Information about the radiological conditions in the plant and its immediate surroundings, and about meteorological conditions in the vicinity of the plant, shall be accessible from the ERF. Guidance The design provides emergency support facilities which include a technical support center and an onsite emergency response facility

	The TSC will provide the following functions: • provide technical support and plant management to plant operation personnel during emergency conditions • handle peripheral duties and communication not directly related to reactor manipulations in order to relieve the burden of reactor operators during emergency conditions • prevent congestion in the control rooms • perform emergency support functions until the emergency response facility is functional To facilitate the above functions, the TSC should be located as close as possible to control rooms with sufficient size to accommodate the technical support staff. Equipment should be provided to gather, store, and display data needed in the TSC to analyze plant conditions. The TSC should have a complete and up-to-date repository of plant records and to aid the technical analysis and evaluation of emergency conditions. Equipment should be provided in the emergency response facility for the acquisition, display, and evaluation of all radiological, meteorological, and plant system data pertinent to determine offsite protective measures. Equipment used in performing essential emergency response facility functions should be located within the emergency response facility complex. However, supplemental calculations and analytical support of emergency response facility evaluations may be provided from facilities outside the emergency response facility. The emergency response facility data system should be designed to achieve an appropriate level of reliability. The location of the emergency response facility should ensure optimum functional and reliability characteristics for carrying out its specific functions. If the TSC and emergency response facility are located in one place, then they should be physically separate from the control rooms with adequate distance to ensure the capability of carrying out its functions. In the case of plants with multiple units at a site, the emergency support facilities should be demonstrated to be adequate to respond to common-cause events in multiple units.
Macro-Gap	SF01-22-16
Issue/Gap Description	The Bruce B design does not provide an onsite emergency facility (or facilities) that are separate from the plant control rooms, which include a

	SPDS similar to those in the MCR and in the SCA. Therefore, there are no design provisions for such a facility (or facilities) to protect occupants from DBA or DEC conditions and be equipped to allow extended operation as required in this clause. This is considered a gap (Gap).
Rationale	Provision of onsite emergency support facilities that are separate from the plant control rooms for use by the technical support staff and emergency support staff in the event of an emergency would require completely new structures to be built at the Bruce site which is only practicable at the overall site layout phase and definition of facilities for a new plant. It should also be noted that current facilities meet the regulatory requirements on emergency response facilities The current Bruce LCH states “Clause 2.2.6(4) of REGDOC-2.10.1 is satisfied by the current location of Bruce Power's Emergency Management Centre with supporting procedures on security and communications arrangements as described in the clause”. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.10.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.10.4 Credit for operator action
Requirement Assessed	If operator action is required for actuation of any safety system or safety support system equipment, all of the following requirements shall apply: 1. there are clear, well-defined, validated, and readily available operating procedures that identify the necessary actions 2. there is instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action 3. following indication of the necessity for operator action inside the control rooms, there are at least 30 minutes available before the operator action is required 4. following indication of the necessity for operator action outside the control rooms, there is a minimum of 1 hour available before the operator action is required For automatically initiated safety systems and control logic actions, the design shall facilitate backup manual initiation from inside the appropriate control room. Guidance The design should ensure that no failure of monitoring or display systems will influence the functioning of other safety systems. The available time before operator action can be credited should be counted from the receipt of an unambiguous indication of a potential accident (typically an alarm) and includes diagnostic time. The time available to perform the actions should be based on the analysis of the plant response to AOOs and DBAs, using realistic assumptions. The time required for operator action should be based on a human factors engineering analysis of operator response time, which (in turn) is based on a documented sequence of operator actions. Uncertainties in the analysis of time required are identified and assessed. An adequate time margin should also be added to the analyzed time. If operator action is required for actuation of any safety function, other than meeting the requirements of this regulatory document, the analysis should also demonstrate that: • there is sufficient time available for the operator to perform the required manual action

	• the operator can perform the actions correctly and reliably in the time available The sequence of actions should use only alarms, controls, and displays that would be available in locations where the tasks will be performed and should be available in all scenarios analysed. A preliminary validation should be conducted, to provide independent confirmation to the validity of the estimated “time available” and “time required” for human actions. The preliminary validation results should support the conclusion that the time required, including margin, to perform individual steps and the overall documented sequence of manual operator actions are reasonable, realistic, repeatable, and bounded by the initial analysis. An integrated system test should also be conducted, to validate the manual actions credited in the safety analysis, using a full-scale simulator. Tasks conducted outside the control room should be included in the integrated system validations. Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field location.
Macro-Gap	SF01-07-15
Issue/Gap Description	Operator actions in Part 3 of the Safety Report are assumed to be 15 minutes for actions inside the control room and 30 minutes for actions outside the control room. These assumptions clearly do not meet the proposed values for new plants but they are consistent with the guidance of REGDOC 2.4.1 and CSA 290.1. Therefore, it is assessed as a gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis which puts operator action times inside the control room at 15 minutes as documented in the Bruce A Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. Changing operator action time from 15 minutes to 30 minutes for actions inside the control room and from 30 minutes to 1 hour for actions outside will require fundamental changes to safety and safety support system design. For example, this would require changes to meet new requirements in terms of SSC actuation, capacity, and performance, etc., which is impracticable. Also as noted in the gap description assumptions used in the safety analysis are consistent with the guidance of REGDOC 2.4.1 and CSA 290.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.10.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.10.4 Credit for operator action
Requirement Assessed	If operator action is required for actuation of any safety system or safety support system equipment, all of the following requirements shall apply: 1. there are clear, well-defined, validated, and readily available operating procedures that identify the necessary actions 2. there is instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action 3. following indication of the necessity for operator action inside the control rooms, there are at least 30 minutes available before the operator action is required 4. following indication of the necessity for operator action outside the control rooms, there is a minimum of 1 hour available before the operator action is required For automatically initiated safety systems and control logic actions, the design shall facilitate backup manual initiation from inside the appropriate control room. Guidance The design should ensure that no failure of monitoring or display systems will influence the functioning of other safety systems. The available time before operator action can be credited should be counted from the receipt of an unambiguous indication of a potential accident (typically an alarm) and includes diagnostic time. The time available to perform the actions should be based on the analysis of the plant response to AOOs and DBAs, using realistic assumptions. The time required for operator action should be based on a human factors engineering analysis of operator response time, which (in turn) is based on a documented sequence of operator actions. Uncertainties in the analysis of time required are identified and assessed. An adequate time margin should also be added to the analyzed time. If operator action is required for actuation of any safety function, other than meeting the requirements of this regulatory document, the analysis should also demonstrate that: • there is sufficient time available for the operator to perform the required manual action

	• the operator can perform the actions correctly and reliably in the time available The sequence of actions should use only alarms, controls, and displays that would be available in locations where the tasks will be performed and should be available in all scenarios analysed. A preliminary validation should be conducted, to provide independent confirmation to the validity of the estimated “time available” and “time required” for human actions. The preliminary validation results should support the conclusion that the time required, including margin, to perform individual steps and the overall documented sequence of manual operator actions are reasonable, realistic, repeatable, and bounded by the initial analysis. An integrated system test should also be conducted, to validate the manual actions credited in the safety analysis, using a full-scale simulator. Tasks conducted outside the control room should be included in the integrated system validations. Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field location.
Macro-Gap	SF01-09-16
Issue/Gap Description	Operator actions in Part 3 of the Safety Report are assumed to be 15 minutes for actions inside the control room and 30 minutes for actions outside the control room. These assumptions clearly do not meet the proposed values for new plants but they are consistent with the guidance of REGDOC 2.4.1 and CSA 290.1. Therefore, it is assessed as a gap (Gap).
Rationale	Bruce Power is in compliance with the current licensing basis which puts operator action times inside the control room at 15 minutes as documented in the Bruce B Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. Changing operator action time from 15 minutes to 30 minutes for actions inside the control room and from 30 minutes to 1 hour for actions outside will require fundamental changes to safety and safety support system design. For example, this would require changes to meet new requirements in terms of SSC actuation, capacity, and performance, etc., which is impracticable. Also as noted in the gap description assumptions used in the safety analysis are consistent with the guidance of REGDOC 2.4.1 and CSA 290.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.12.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.12.2 Handling and storage of irradiated fuel
Requirement Assessed	The design of the handling and storage systems for irradiated fuel shall: 1. ensure nuclear criticality safety 2. permit adequate heat removal in operational states, DBAs and DECs 3. permit inspection of irradiated fuel 4. permit periodic inspection and testing of components important to safety 5. prevent the dropping of irradiated fuel in transit 6. prevent unacceptable handling stresses on fuel elements or fuel assemblies 7. prevent the inadvertent dropping of heavy objects and equipment on fuel assemblies 8. permit inspection and safe storage of suspect or damaged fuel elements or fuel assemblies 9. provide proper means for radiation protection 10. permit adequate identification of individual fuel modules 11. facilitate maintenance and decommissioning of the fuel storage and handling facilities 12. facilitate decontamination of fuel handling and storage areas and equipment when necessary 13. ensure implementation of adequate operating and accounting procedures to prevent loss of fuel 14. include measures to prevent a direct threat or sabotage to irradiated fuel 15. meet Canada’s safeguards requirements for recording and reporting accountancy data, and for monitoring flows and inventories related to irradiated fuel containing fissile material A design for a water pool used for fuel storage shall include provisions for:

	1. controlling the chemistry and activity of any water in which irradiated fuel is handled or stored 2. monitoring and controlling the water level in the fuel storage pool 3. detecting leakage 4. preventing the pool from emptying in the event of a pipe break 5. sufficient space to accommodate the entire reactor core inventory at all times The design of irradiated fuel storage pools shall include means for preventing the uncovering of fuel in the pool in operational states, DBAs and DECs. The design for a water pool used for fuel storage shall include provisions for DECs by: 1. ensuring that boiling in the pool does not result in structural damage 2. providing temporary connections to enable the refill of the pool using temporary supplies 3. providing temporary connections to heat removal systems for power and cooling water 4. providing hydrogen mitigation in the spent fuel pool area 5. ensuring that severe accident management actions related to the spent fuel pool can be carried out Guidance Hydrogen mitigation in the spent fuel pool area is particularly important if it is envisaged that the pool may be used for fission product scrubbing as part of containment venting. Hydrogen mitigation in the spent fuel pool area may not be necessary if draining of the pool beyond make- up capability can be precluded.
Macro-Gap	SF01-13-15
Issue/Gap Description	The requirement for sufficient space to accommodate the entire reactor core inventory at all times is not reflected in the design and operating documentation. Therefore, it is assessed as a gap (Gap). The Used Fuel Waste and Cobalt 60 Agreement defines the Buffer Capacity and discusses the required capacity of Used Fuel Pools in respect of either the Bruce A or Bruce B. The Used Fuel Pools should be sufficient to hold one reactor core dump plus the amount of used fuel waste reasonably projected by Bruce Power to be generated during one year by the number of operational Bruce A reactors or Bruce B reactors associated with such

	used fuel pools. The term Used Fuel Pools does not include the primary water pools associated with Bruce A or Bruce B.
Rationale	Bruce A meets the current licensing basis which requires space availability for one reactor core inventory at all times. Bruce A and Bruce B used fuel pools meet the requirements of Used Fuel Waste and Cobalt 60 Agreement which defines the Buffer Capacity. This requirement for a new plant would impose construction of an additional IFB for each unit which is impracticable. Bruce Power would be able to manage through alternative means if a situation arose such that the cores of all four units were required to be defueled. This would involve transferring existing inventory in the Secondary Irradiated Fuel Bays to dry storage to make storage space available in the irradiated fuel bays. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.12.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.12.2 Handling and storage of irradiated fuel
Requirement Assessed	The design of the handling and storage systems for irradiated fuel shall: 1. ensure nuclear criticality safety 2. permit adequate heat removal in operational states, DBAs and DECs 3. permit inspection of irradiated fuel 4. permit periodic inspection and testing of components important to safety 5. prevent the dropping of irradiated fuel in transit 6. prevent unacceptable handling stresses on fuel elements or fuel assemblies 7. prevent the inadvertent dropping of heavy objects and equipment on fuel assemblies 8. permit inspection and safe storage of suspect or damaged fuel elements or fuel assemblies 9. provide proper means for radiation protection 10. permit adequate identification of individual fuel modules 11. facilitate maintenance and decommissioning of the fuel storage and handling facilities 12. facilitate decontamination of fuel handling and storage areas and equipment when necessary 13. ensure implementation of adequate operating and accounting procedures to prevent loss of fuel 14. include measures to prevent a direct threat or sabotage to irradiated fuel 15. meet Canada's safeguards requirements for recording and reporting accountancy data, and for monitoring flows and inventories related to irradiated fuel containing fissile material A design for a water pool used for fuel storage shall include provisions for:

	1. controlling the chemistry and activity of any water in which irradiated fuel is handled or stored 2. monitoring and controlling the water level in the fuel storage pool 3. detecting leakage 4. preventing the pool from emptying in the event of a pipe break 5. sufficient space to accommodate the entire reactor core inventory at all times The design of irradiated fuel storage pools shall include means for preventing the uncovering of fuel in the pool in operational states, DBAs and DECs. The design for a water pool used for fuel storage shall include provisions for DECs by: 1. ensuring that boiling in the pool does not result in structural damage 2. providing temporary connections to enable the refill of the pool using temporary supplies 3. providing temporary connections to heat removal systems for power and cooling water 4. providing hydrogen mitigation in the spent fuel pool area 5. ensuring that severe accident management actions related to the spent fuel pool can be carried out Guidance Hydrogen mitigation in the spent fuel pool area is particularly important if it is envisaged that the pool may be used for fission product scrubbing as part of containment venting. Hydrogen mitigation in the spent fuel pool area may not be necessary if draining of the pool beyond make- up capability can be precluded.
Macro-Gap	SF01-13-16
Issue/Gap Description	The requirement for sufficient space to accommodate the entire reactor core inventory at all times is not reflected in the design and operating documentation. Therefore, it is assessed as a gap (Gap).
Rationale	Bruce B meets the current licensing basis which requires space availability for one reactor core inventory at all times. Bruce A and Bruce B used fuel pools meet the requirements of Used Fuel Waste and Cobalt 60 Agreement which defines the Buffer Capacity. This requirement for a new nuclear plant would impose construction of an additional IFB for each unit

	which is impracticable. Bruce Power would be able to manage through alternative means if a situation arose such that the cores of all four units were required to be defueled. This would involve transferring existing inventory in the Secondary Irradiated Fuel Bays to dry storage to make storage space available in the irradiated fuel bays. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.3.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.3.2 Steam and feedwater system piping and vessels
Requirement Assessed	All piping and vessels shall be typically separated from electrical and control systems, to the extent practicable. The auxiliary feedwater, steam generator pressure control, and other auxiliary systems, shall prevent the escalation of AOOs to DBAs or DECs.
Macro-Gap	SF01-01-15
Issue/Gap Description	A systematic review of the design of auxiliary feedwater, steam generator pressure control, and other auxiliary systems has not been performed to demonstrate that they would prevent the escalation of AOOs to accident conditions. Therefore, this is assessed as a gap (Gap). The topic of AOOs is addressed in detail under Safety Factor 5.
Rationale	Original design of SSCs was not based on the same definition of event classes. Specifically design of auxiliary feedwater, steam generator pressure control, and other auxiliary systems, did not include the requirement to prevent the escalation of AOOs to DBAs or DECs. Such a requirement is considered to be specifically applicable to a new nuclear plant. This would require changing the design basis of auxiliary feedwater, steam generator pressure control, and other auxiliary systems and may also impact the design of reactor systems and hence is impracticable. AOOs will be addressed as part of AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.3.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.3.2 Steam and feedwater system piping and vessels
Requirement Assessed	All piping and vessels shall be typically separated from electrical and control systems, to the extent practicable. The auxiliary feedwater, steam generator pressure control, and other auxiliary systems, shall prevent the escalation of AOOs to DBAs or DECs.
Macro-Gap	SF01-01-16
Issue/Gap Description	A systematic review of the design of auxiliary feedwater, steam generator pressure control, and other auxiliary systems has not been performed to demonstrate that they would prevent the escalation of AOOs to accident conditions. Therefore, this is assessed as a gap (Gap). The topic of AOOs is addressed in detail under Safety Factor 5.
Rationale	Original design of SSCs was not based on the same definition of event classes. Specifically design of auxiliary feedwater, steam generator pressure control, and other auxiliary systems, did not include the requirement to prevent the escalation of AOOs to DBAs or DECs. Such a requirement is considered to be specifically applicable to a new nuclear plant. This would require changing the design basis of auxiliary feedwater, steam generator pressure control, and other auxiliary systems and may also impact the design of reactor systems and hence is impracticable. AOOs will be addressed as part of AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.4.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.4.2 Reliability
Requirement Assessed	The design shall permit ongoing demonstration that each means of shutdown is being operated and maintained in a manner that ensures continued adherence to reliability and effectiveness requirements. Periodic testing of the systems and their components shall be scheduled at a frequency commensurate with applicable requirements. Guidance The reliability calculation should include sensing the need for shutdown, initiation of shutdown, and insertion of negative reactivity. All elements necessary to complete the shutdown function should be included. The reliability of the shutdown function should be such that the cumulative frequency of failure to shutdown on demand is less than 1E-5 failures per demand, and the contribution of all sequences involving failure to shutdown to the large release frequency of the safety goals is less than 1E-7/yr. This considers the likelihood of the initiating event and recognizes that the two shutdown means may not be completely independent. Section 7.6.2 requires that the shutdown function be delivered even in the presence of any single failure and even during the worst configuration from testing and maintenance. For example, for a rod based system to meet the SFC, the safety analysis may assume that the two highest worth control rods are unavailable (one for testing, and one assumed to fail on demand, in accordance with the SFC). In this case, no further testing of rods would be allowed until the rod under testing becomes available.
Macro-Gap	SF01-05-15
Issue/Gap Description	Results of the Level 2 Internal Events At-Power PSA NK21-03611.5 P NSAS Ver1 indicate that the contribution to the large release frequency from all sequences involving failure to shutdown is about 2.3 E-7 occurrences per reactor per year. Accordingly, the proposed safety goal of 1E-7/yr is not met, which constitutes a gap with respect to the guidance portion of this clause (Gap).
Rationale	Bruce A is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement. Such a requirement is considered to be specifically applicable to a new nuclear plant. Reducing the frequency of failure to shut down below 1E-5 to meet the LRF limit of 1E-7 would require fundamental design changes to the shutdown systems which are impracticable. This micro-gap has been judged to have low safety significance, and

	would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.6.12_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.6.12 Design extension conditions
Requirement Assessed	Following onset of core damage, the containment boundary shall be capable of contributing to the reduction of radioactivity releases to allow sufficient time for the implementation of offsite emergency procedures. Damage to the containment structure shall be limited to prevent uncontrolled releases of radioactivity, and to maintain the integrity of structures that support internal components. The ability of the containment system to withstand loads associated with design extension conditions (DECs) shall be demonstrated in design documentation, and shall include the following considerations: 1. various heat sources, including residual heat, metal-water reactions, combustion of gases, and standing flames 2. pressure control 3. control of combustible gases 4. sources of non-condensable gases 5. control of radioactive material leakage 6. effectiveness of isolation devices 7. functionality and leak tightness of airlocks and containment penetrations 8. effects of the accident on the integrity and functionality of internal structures The design authority shall demonstrate that complementary design features have been incorporated that will: 1. prevent a containment melt-through or failure due to the thermal impact of the core debris 2. facilitate cooling of the core debris 3. minimize generation of non-condensable gases and radioactive products 4. preclude unfiltered and uncontrolled release from containment

	Guidance Provisions for DECs vary greatly between designs. The claimed functionality and analysis should be supported by adequate evidence. The containment leakage rate in DECs with core damage should not exceed the design leakage rate for a sufficient period to allow for the implementation of offsite emergency measures. This period should be demonstrated, with reasonable confidence, to be at least 24 hours. The design should minimize generation of combustible, non-condensable gases from corium- concrete interaction. Containment venting design should take into account such factors as: • ignition of flammable gases • generation of non-condensable gases • impact on filters by containment environmental conditions, such as radioactive materials, high temperature and high humidity Experimental or analytical evidence should be provided to demonstrate that venting will not lead to unfiltered and uncontrolled releases of radioactive materials into the environment.
Macro-Gap	SF01-11-15
Issue/Gap Description	A review of the same clause in RD-337 indicated that the Bruce A design does not fully meet this requirement, as documented in [NK21-CORR-00531-11005]. Bruce A containment has been shown capable of withstanding the conditions of severe accidents such that the leakage requirements are met. The consequences of the aspects of severe accidents listed in this clause are mitigated by SAMG, as discussed earlier. The current design documentation does not explicitly consider the load conditions during DECs. Therefore, it is assessed as a gap. (Gap 1)
Rationale	Bruce A meets the current licensing basis as documented in the Bruce A Safety Report. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Bruce A containment has been shown capable of withstanding the conditions of severe accidents such that the leakage requirements are met. The consequences of the aspects of severe accidents listed in this clause are mitigated by SAMG. Such a requirement is considered to be specifically applicable to a new nuclear plant. Original containment design did not explicitly consider ability of the containment system to withstand loads associated with DECs and

	particularly those associated with onset of core damage. Compliance would require fundamental design changes to containment which is impracticable. Onset of core damage is considered to be a condition associated with a severe accident and as such mitigation of consequences of such an event is covered by the SAMG program. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.6.12_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.6.12 Design extension conditions
Requirement Assessed	Following onset of core damage, the containment boundary shall be capable of contributing to the reduction of radioactivity releases to allow sufficient time for the implementation of offsite emergency procedures. Damage to the containment structure shall be limited to prevent uncontrolled releases of radioactivity, and to maintain the integrity of structures that support internal components. The ability of the containment system to withstand loads associated with design extension conditions (DECs) shall be demonstrated in design documentation, and shall include the following considerations: 1. various heat sources, including residual heat, metal-water reactions, combustion of gases, and standing flames 2. pressure control 3. control of combustible gases 4. sources of non-condensable gases 5. control of radioactive material leakage 6. effectiveness of isolation devices 7. functionality and leak tightness of airlocks and containment penetrations 8. effects of the accident on the integrity and functionality of internal structures The design authority shall demonstrate that complementary design features have been incorporated that will: 1. prevent a containment melt-through or failure due to the thermal impact of the core debris 2. facilitate cooling of the core debris 3. minimize generation of non-condensable gases and radioactive products 4. preclude unfiltered and uncontrolled release from containment

	Guidance Provisions for DECs vary greatly between designs. The claimed functionality and analysis should be supported by adequate evidence. The containment leakage rate in DECs with core damage should not exceed the design leakage rate for a sufficient period to allow for the implementation of offsite emergency measures. This period should be demonstrated, with reasonable confidence, to be at least 24 hours. The design should minimize generation of combustible, non-condensable gases from corium- concrete interaction. Containment venting design should take into account such factors as: • ignition of flammable gases • generation of non-condensable gases • impact on filters by containment environmental conditions, such as radioactive materials, high temperature and high humidity Experimental or analytical evidence should be provided to demonstrate that venting will not lead to unfiltered and uncontrolled releases of radioactive materials into the environment.
Macro-Gap	SF01-01-15
Issue/Gap Description	The Bruce A containment floor was not constructed with concrete used to limit the production of non-condensable gases due to core-concrete interactions. However, the concrete containment floor and, in particular, the fueling vault floor, will withstand corium concrete interaction assuming that there is sufficient water on containment floor and sufficient floor surface area for corium relocation. Bruce Power is evaluating various options for longer-term provisions to ensure core cooling and In Vessel Retention (IVR) of corium debris in the event that an accident has progressed to a severe accident. These options include makeup water to the calandria, PHTS and shield tank. Bruce Power will continue to review the benefits of and the need for the installation of additional provisions for core cooling using emergency makeup provisions to the heat transport, moderator and shield cooling systems. The review will consider the Bruce specific analysis described above, the results of the CANDU Owner's Group(COG) post-Fukushima Joint Project (JR 4426) review of IVR, and the results of a Risk Informed Decision Making (RIDM) -based assessment. The original design did not include complementary design features for severe accidents; therefore this is assessed as a gap. (Gap 2)
Rationale	Bruce A plant design basis, documented in the Bruce A Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. Original containment design did not explicitly consider DECs and particularly mitigation features due to the consequences of onset of

	core damage. Fundamental design changes to containment would be required to incorporate mitigation features to limit non-condensable gasses due to the consequences of onset of core damage which is impracticable. Onset of core damage is considered to be a condition associated with a severe accident and as such is covered by the SAMG program. The scope of the work in IIP-2014, GIO-011 Implement Enhancements to SAMG, includes the following as related to this gap: • Assessment of plant habitability under severe accident conditions and identification of modifications required. • Improvement to understanding of severe accident phenomena including containment Integrity, hydrogen production, aerosol behaviour, and in-vessel retention. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits. Reference: NK21-CORR-00531-11567 SIP-11: Tracking is as follows: (1) SAMG to include multi-unit events = FAI 3.1.2 and RegM 28415639; (2) equipment survivability = FAI 1.8.1 and RegM 28415592; (3) habitability = FAI 1.9.1 and RegM 28415634

Gap #	SF01_CNSC REGDOC 2.5.2_8.6.12_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.6.12 Design extension conditions
Requirement Assessed	Following onset of core damage, the containment boundary shall be capable of contributing to the reduction of radioactivity releases to allow sufficient time for the implementation of offsite emergency procedures. Damage to the containment structure shall be limited to prevent uncontrolled releases of radioactivity, and to maintain the integrity of structures that support internal components. The ability of the containment system to withstand loads associated with design extension conditions (DECs) shall be demonstrated in design documentation, and shall include the following considerations: 1. various heat sources, including residual heat, metal-water reactions, combustion of gases, and standing flames 2. pressure control 3. control of combustible gases 4. sources of non-condensable gases 5. control of radioactive material leakage 6. effectiveness of isolation devices 7. functionality and leak tightness of airlocks and containment penetrations 8. effects of the accident on the integrity and functionality of internal structures The design authority shall demonstrate that complementary design features have been incorporated that will: 1. prevent a containment melt-through or failure due to the thermal impact of the core debris 2. facilitate cooling of the core debris 3. minimize generation of non-condensable gases and radioactive products 4. preclude unfiltered and uncontrolled release from containment

	Guidance Provisions for DECs vary greatly between designs. The claimed functionality and analysis should be supported by adequate evidence. The containment leakage rate in DECs with core damage should not exceed the design leakage rate for a sufficient period to allow for the implementation of offsite emergency measures. This period should be demonstrated, with reasonable confidence, to be at least 24 hours. The design should minimize generation of combustible, non-condensable gases from corium- concrete interaction. Containment venting design should take into account such factors as: • ignition of flammable gases • generation of non-condensable gases • impact on filters by containment environmental conditions, such as radioactive materials, high temperature and high humidity Experimental or analytical evidence should be provided to demonstrate that venting will not lead to unfiltered and uncontrolled releases of radioactive materials into the environment.
Macro-Gap	SF01-11-16
Issue/Gap Description	Bruce B containment has been shown capable of withstanding the conditions of severe accidents such that the leakage requirements are met. The consequences of the aspects of severe accidents listed in this clause are mitigated by SAMG, as discussed earlier. The current design documentation does not explicitly consider the load conditions during DECs. Therefore, it is assessed as a gap. (Gap 1)
Rationale	Bruce B meets the current licensing basis as documented in the Bruce B Safety Report. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Bruce B containment has been shown capable of withstanding the conditions of severe accidents such that the leakage requirements are met. The consequences of the aspects of severe accidents listed in this clause are mitigated by SAMG. Such a requirement is considered to be specifically applicable to a new nuclear plant. Original containment design did not explicitly consider ability of the containment system to withstand loads associated with DECs and particularly those associated with onset of core damage. Compliance would require fundamental design changes to containment which is

	impracticable. Onset of core damage is considered to be a condition associated with a severe accident and as such mitigation of consequences of such an event is covered by the SAMG program. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.8_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.8 Emergency heat removal system
Requirement Assessed	The design shall include an emergency heat removal system (EHRS) which provides for removal of residual heat in order to meet fuel design limits and reactor coolant boundary condition limits. If the design of the plant is such that the EHRS is required to mitigate the consequences of a DBA, then the EHRS shall be designed as a safety system. There shall be reasonable confidence that the EHRS will function during DECs, if required. Correct operation of the EHRS equipment following an accident shall not be dependent on power supplies from the electrical grid or from the turbine generators associated with any reactor unit that is located on the same site as the reactor involved in the accident. Where water is required for the EHRS, it shall come from a source that is independent of normal supplies. The design shall support maintenance and reliability testing without a reduction in system effectiveness below what is required by the OLCs. As far as practicable, inadvertent operation of the EHRS, or of part of the EHRS, shall not have a detrimental effect on plant safety. If the fire water supply or system components are interconnected to the EHRS, operation of one shall not impair operation of the other. Guidance The emergency heat removal system is to provide a path to ultimate heat sink, in the case that normal heat removal capabilities are not available. The purpose of this system is to prevent events from escalating and to mitigate their consequences. Emergency heat removal relates to post-accident heat removal and may be provided by a number of systems, depending on circumstances: • post-LOCA heat removal may be provided by ECCS (refer to section 8.5) • for non-LOCA events, emergency heat removal may be through primary or secondary cooling systems For all means of emergency heat removal, the design should be such that all equipment is appropriately designed to function in the class of accidents for which it is credited.

	If the system credited has another role in normal operation, then the design should be such that the system will meet the requirements of a safety system when used in DBAs or DECs. The design basis requirements for the system in this role should be provided. Many of the actions associated with operation of the systems credited for emergency heat removal may not be initiated automatically. When there is reliance on manual operation, the review of human factors considerations should have very high importance. Primary side emergency heat removal could be through normal shutdown cooling means. The design should be such that: • a means of depressurizing the primary system is provided and the means of depressurization meets the requirements of a safety system, or • the shutdown cooling system is capable of being operated at full primary pressure and temperature Passive or non-passive (e.g., natural circulation or pumped) heat removal may be used. Non- passive systems require emergency power. Natural circulation systems should demonstrate the capability over the full range of applicable operating conditions. Secondary side emergency heat removal that relies on water being provided to the secondary side of steam generators may be provided by a separate pumped supply or by a secondary depressurization and gravity feed. The water supply should meet the requirements of a safety system.
Macro-Gap	SF01-11-15
Issue/Gap Description	As documented in [RABA 0804].Bruce A design does not provide this fifth (special) safety system, as these requirements were intended for new build NPPs. For Bruce A, the emergency heat removal function is provided by Emergency Boiler Cooling, Shutdown Cooling, and Maintenance Cooling Systems. The emergency heat removal function is provided by the Emergency Water System, the Shutdown Cooling System and the Maintenance Cooling System. A redundancy and diversity assessment of these systems was performed for Bruce 1&2, and it was concluded that changes to plant design and procedures are not warranted. The Emergency Heat Removal function is provided by more than one system; hence there are several ways this cool down could take place. Since the emergency heat removal function is provided by more than one system; it cannot be confirmed that the same function will be available during DECs, if required. Therefore, this is assessed as a gap (Gap).
Rationale	Bruce A meets the current licensing basis as documented in the Bruce A Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. An additional safety system design and installation within the current plant configuration is impracticable.

	This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits. As described in the assessment the Emergency Heat Removal function is provided by more than one system; hence there are several ways this cooldown could take place. Bruce Power has also added Emergency Mitigating Equipment to allow addition of water into the Steam Generators (complete), Heat Transport System (in progress), the Moderator (Complete Units 1 & 2, in progress Units 3 & 4) and the Shield Tank (in progress) to provide additional heat removal capability in the case that the normal and back up heat sinks become unavailable.

Gap #	SF01_CNSC REGDOC 2.5.2_8.8_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.8 Emergency heat removal system
Requirement Assessed	The design shall include an emergency heat removal system (EHRS) which provides for removal of residual heat in order to meet fuel design limits and reactor coolant boundary condition limits. If the design of the plant is such that the EHRS is required to mitigate the consequences of a DBA, then the EHRS shall be designed as a safety system. There shall be reasonable confidence that the EHRS will function during DECs, if required. Correct operation of the EHRS equipment following an accident shall not be dependent on power supplies from the electrical grid or from the turbine generators associated with any reactor unit that is located on the same site as the reactor involved in the accident. Where water is required for the EHRS, it shall come from a source that is independent of normal supplies. The design shall support maintenance and reliability testing without a reduction in system effectiveness below what is required by the OLCs. As far as practicable, inadvertent operation of the EHRS, or of part of the EHRS, shall not have a detrimental effect on plant safety. If the fire water supply or system components are interconnected to the EHRS, operation of one shall not impair operation of the other. Guidance The emergency heat removal system is to provide a path to ultimate heat sink, in the case that normal heat removal capabilities are not available. The purpose of this system is to prevent events from escalating and to mitigate their consequences. Emergency heat removal relates to post-accident heat removal and may be provided by a number of systems, depending on circumstances: • post-LOCA heat removal may be provided by ECCS (refer to section 8.5) • for non-LOCA events, emergency heat removal may be through primary or secondary cooling systems For all means of emergency heat removal, the design should be such that all equipment is appropriately designed to function in the class of accidents for which it is credited.

	If the system credited has another role in normal operation, then the design should be such that the system will meet the requirements of a safety system when used in DBAs or DECs. The design basis requirements for the system in this role should be provided. Many of the actions associated with operation of the systems credited for emergency heat removal may not be initiated automatically. When there is reliance on manual operation, the review of human factors considerations should have very high importance. Primary side emergency heat removal could be through normal shutdown cooling means. The design should be such that: • a means of depressurizing the primary system is provided and the means of depressurization meets the requirements of a safety system, or • the shutdown cooling system is capable of being operated at full primary pressure and temperature Passive or non-passive (e.g., natural circulation or pumped) heat removal may be used. Non- passive systems require emergency power. Natural circulation systems should demonstrate the capability over the full range of applicable operating conditions. Secondary side emergency heat removal that relies on water being provided to the secondary side of steam generators may be provided by a separate pumped supply or by a secondary depressurization and gravity feed. The water supply should meet the requirements of a safety system.
Macro-Gap	SF01-23-16
Issue/Gap Description	Bruce A and B design does not provide this fifth (special) safety system, as these requirements were intended for new build NPPs. For Bruce B the emergency heat removal function is provided by the Emergency Water System, the Shutdown Cooling System and the Maintenance Cooling System (Gap 1).
Rationale	Bruce B meets the current licensing basis as documented in the Bruce B Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. An additional safety system design and installation within the current plant configuration is impracticable. As described in the assessment the Emergency Heat Removal function is provided by more than one system; hence there are several ways this cooldown could take place. Bruce Power has also added Emergency Mitigating Equipment to allow addition of water into the Steam Generators (complete), Heat Transport System (in progress), the Moderator (in progress) and the Shield Tank (in progress) to provide additional heat removal capability in the case that the normal and back up heat sinks become unavailable.

	This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CNSC REGDOC 2.5.2_8.8_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.8 Emergency heat removal system
Requirement Assessed	The design shall include an emergency heat removal system (EHRS) which provides for removal of residual heat in order to meet fuel design limits and reactor coolant boundary condition limits. If the design of the plant is such that the EHRS is required to mitigate the consequences of a DBA, then the EHRS shall be designed as a safety system. There shall be reasonable confidence that the EHRS will function during DECs, if required. Correct operation of the EHRS equipment following an accident shall not be dependent on power supplies from the electrical grid or from the turbine generators associated with any reactor unit that is located on the same site as the reactor involved in the accident. Where water is required for the EHRS, it shall come from a source that is independent of normal supplies. The design shall support maintenance and reliability testing without a reduction in system effectiveness below what is required by the OLCs. As far as practicable, inadvertent operation of the EHRS, or of part of the EHRS, shall not have a detrimental effect on plant safety. If the fire water supply or system components are interconnected to the EHRS, operation of one shall not impair operation of the other. Guidance The emergency heat removal system is to provide a path to ultimate heat sink, in the case that normal heat removal capabilities are not available. The purpose of this system is to prevent events from escalating and to mitigate their consequences. Emergency heat removal relates to post-accident heat removal and may be provided by a number of systems, depending on circumstances: • post-LOCA heat removal may be provided by ECCS (refer to section 8.5) • for non-LOCA events, emergency heat removal may be through primary or secondary cooling systems For all means of emergency heat removal, the design should be such that all equipment is appropriately designed to function in the class of accidents for which it is credited.

	If the system credited has another role in normal operation, then the design should be such that the system will meet the requirements of a safety system when used in DBAs or DECs. The design basis requirements for the system in this role should be provided. Many of the actions associated with operation of the systems credited for emergency heat removal may not be initiated automatically. When there is reliance on manual operation, the review of human factors considerations should have very high importance. Primary side emergency heat removal could be through normal shutdown cooling means. The design should be such that: • a means of depressurizing the primary system is provided and the means of depressurization meets the requirements of a safety system, or • the shutdown cooling system is capable of being operated at full primary pressure and temperature Passive or non-passive (e.g., natural circulation or pumped) heat removal may be used. Non- passive systems require emergency power. Natural circulation systems should demonstrate the capability over the full range of applicable operating conditions. Secondary side emergency heat removal that relies on water being provided to the secondary side of steam generators may be provided by a separate pumped supply or by a secondary depressurization and gravity feed. The water supply should meet the requirements of a safety system.
Macro-Gap	SF01-23-16
Issue/Gap Description	Since the emergency heat removal function is provided by more than one system; it cannot be confirmed that the same function will be available during DECs, if required. Therefore, this is assessed as a gap (Gap 2).
Rationale	Bruce B meets the current licensing basis as documented in the Bruce B Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. An additional safety system design and installation within the current plant configuration is impracticable. As described in the assessment the Emergency Heat Removal function is provided by more than one system; hence there are several ways this cooldown could take place. Bruce Power has also added Emergency Mitigating Equipment to allow addition of water into the Steam Generators (complete), Heat Transport System (in progress), the Moderator (in progress) and the Shield Tank (in progress) to provide additional heat removal capability in the case that the normal and back up heat sinks become unavailable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.1-4.4_15
Document ID	CSA N290.0-11
Article/Clause	4.1-4.4
Requirement Assessed	Define general requirements related to the plant states and system operating states.
Macro-Gap	SF01-01-15
Issue/Gap Description	The plant states as defined in clause 4.2 are not explicitly covered in the existing design documentation.
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce A Safety Report. Safety systems at both Bruce A and Bruce B were designed based on single and dual failure considerations which cover the plant states defined in Clause 4.2. However, establishing the design basis in recognition of these states may require fundamental changes at the system and component level which is impracticable. Note that assessment of the plant states under the same classification as in Clause 4.2 is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.1-4.4_16
Document ID	CSA N290.0-11
Article/Clause	4.1-4.4
Requirement Assessed	Define general requirements related to the plant states and system operating states.
Macro-Gap	SF01-01-16
Issue/Gap Description	Gap against clause 4.2: The plant states as defined in clause 4.2 are not explicitly covered in the existing design documentation.
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce B Safety Report. Safety systems at both Bruce A and Bruce B were designed based on single and dual failure considerations which cover the plant states defined in Clause 4.2. However, establishing the design basis in recognition of these states may require fundamental changes at the system and component level which is impracticable. Note that assessment of the plant states under the same classification as in Clause 4.2 is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.5-4.8_15
Document ID	CSA N290.0-11
Article/Clause	4.5-4.8
Requirement Assessed	Present the requirements related to reliability, separation and independence, single failure criteria application and fail-safe design concept.
Macro-Gap	SF01-05-15
Issue/Gap Description	Gap against clause 4.7: The application of the single failure criterion for the Bruce A design does not follow the newer, more restrictive, interpretations of the single failure criterion.
Rationale	Bruce Power is in compliance with the current design basis which reflects the interpretation of single failure criterion that has been accepted in the current licensing basis, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. Application of the newer and more restrictive interpretations of the single failure criterion would lead to fundamental design changes both at the system and component level which is impracticable. Note that assessment of single failures is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.5-4.8_15
Document ID	CSA N290.0-11
Article/Clause	4.5-4.8
Requirement Assessed	Present the requirements related to reliability, separation and independence, single failure criteria application and fail-safe design concept.
Macro-Gap	SF01-01-15
Issue/Gap Description	Gap against clause 4.8: Not all special safety system components are designed such that the most likely failure modes are in the fail-safe direction.
Rationale	Both Bruce A and Bruce B special safety system components were originally designed based on the 'fail safe' principle. New and more restrictive interpretation of this requirement may require design changes at the system and component level which is impracticable. Note that assessment of all special safety system failures is demonstrated in the safety analysis and already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.5-4.8_16
Document ID	CSA N290.0-11
Article/Clause	4.5-4.8
Requirement Assessed	Present the requirements related to reliability, separation and independence, single failure criteria application and fail-safe design concept.
Macro-Gap	SF01-05-16
Issue/Gap Description	Gap against clause 4.7: The application of the single failure criterion for the Bruce A design does not follow the newer, more restrictive, interpretations of the single failure criterion.
Rationale	Bruce Power is in compliance with the current design basis which reflects the interpretation of single failure criterion that has been accepted in the current licensing basis, where licensing requirements imposed only that no single failure in the safety systems should impair their operation. Application of the newer and more restrictive interpretations of the single failure criterion would lead to fundamental design changes both at the system and component level which is impracticable. Note that assessment of single failures is already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.5-4.8_16
Document ID	CSA N290.0-11
Article/Clause	4.5-4.8
Requirement Assessed	Present the requirements related to reliability, separation and independence, single failure criteria application and fail-safe design concept.
Macro-Gap	SF01-20-16
Issue/Gap Description	Gap against clause 4.8: Not all special safety system components are designed such that the most likely failure modes are in the fail-safe direction.
Rationale	Both Bruce A and Bruce B special safety system components were originally designed based on the 'fail safe' principle. New and more restrictive interpretation of this requirement may require design changes at the system and component level which is impracticable. Note that assessment of all special safety system failures is demonstrated in the safety analysis and already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.9-4.13_15
Document ID	CSA N290.0-11
Article/Clause	4.9-4.13
Requirement Assessed	Address the requirements related to safety support systems, pressure-retaining SSCs, instrumentation, control and monitoring, equipment qualification and dynamic piping effects.
Macro-Gap	SF01-01-15
Issue/Gap Description	Gap against clause 4.12.4: The SSCs credited are required to perform their functions during AOOs, DBAs and BDBAs are protected against debris and contaminants initiated by that event and are assessed for their potential to perform under the expected environmental conditions. However, the current design documentation does not consider internal events as leading to AOOs, DBAs and DECs.
Rationale	Bruce A plant design basis, documented in the Bruce A Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not practicable to make wholesale design changes to Civil Structures to protect against internal events and specifically those associated with DECs. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. as those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-019 Assess and improve seismic margins GIO-087 Bruce A Fire Protection Upgrades to Align with CSA-N293-14 (complete) In addition, under GAI 06G01: “Emergency Core Coolant Strainer Deposits” Bruce Power and its Industry Partners addressed concerns about the formation of deposits on ECC system strainers under post-LOCA conditions. It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.9-4.13_16
Document ID	CSA N290.0-11
Article/Clause	4.9-4.13
Requirement Assessed	Address the requirements related to safety support systems, pressure-retaining SSCs, instrumentation, control and monitoring, equipment qualification and dynamic piping effects.
Macro-Gap	SF01-01-16
Issue/Gap Description	Gap against clause 4.12.4: The SSCs credited are required to perform their functions during AOOs, DBAs and BDBAs are protected against debris and contaminants initiated by that event and are assessed for their potential to perform under the expected environmental conditions. However, the current design documentation does not consider internal events as leading to AOOs, DBAs and DECs.
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not practicable to make wholesale design changes to Civil Structures to protect against internal events and specifically those associated with DECs. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. as those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-003 Assess pipe whip and jet impingement GIO-019 Assess and improve seismic margins GIO-092 Bruce B Fire Protection Upgrades to Align with CSA-N293-14 In addition, under GAI 06G01: “Emergency Core Coolant Strainer Deposits” Bruce Power and its Industry Partners addressed concerns about the formation of deposits on ECC system strainers under post-LOCA conditions. It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.0-11_4.9-4.13_16
Document ID	CSA N290.0-11
Article/Clause	4.9-4.13
Requirement Assessed	Address the requirements related to safety support systems, pressure-retaining SSCs, instrumentation, control and monitoring, equipment qualification and dynamic piping effects.
Macro-Gap	SF01-01-16
Issue/Gap Description	Gap against clause 4.12.5: The SSCs credited are required to perform their functions during AOOs, DBAs and BDBAs are protected against debris and contaminants initiated by that event and are assessed for their potential to perform under the expected environmental conditions. However, the current design documentation does not consider internal events as leading to AOOs, DBAs and DECs.
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Such a requirement is considered to be specifically applicable to a new nuclear plant. As a general requirement it is not practicable to make wholesale design changes to Civil Structures to protect against internal events and specifically those associated with DECs. However, engineering and safety analyses are being conducted to assess probability and effects of hazards, e.g. as those related to fire safety, seismic qualification, pipe whip and jet impingement and if required and where practicable, a number of initiatives are being implemented. The current IIP includes the following relevant initiatives: GIO-003 Assess pipe whip and jet impingement GIO-019 Assess and improve seismic margins GIO-092 Bruce B Fire Protection Upgrades to Align with CSA-N293-14 In addition, under GAI 06G01: “Emergency Core Coolant Strainer Deposits” Bruce Power and its Industry Partners addressed concerns about the formation of deposits on ECC system strainers under post-LOCA conditions. It should also be noted that classification and effects of such hazards will be addressed under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.1_4.2.6_16
Document ID	CSA N290.1
Article/Clause	4.2.6 Fail-safe design
Requirement Assessed	The design should aim for fail-safe operation of its SSCs where such an option exists, while maintaining a balance with simplicity. Note: The requirement for fail-safe operation appears in CSA N290.0, Clause 4.8.
Macro-Gap	SF01-20-16
Issue/Gap Description	There remains instances where the failure mode is unsafe and SDS reliability is dependent on panel monitoring or testing by the Operator. For example the ion chamber log N and log rate signals fail unsafe for loss of polarizing voltage (See section 5.17(5) of [CMT-60544-00003 Rev.002]) (Gap 1).
Rationale	Both Bruce A and Bruce B special safety system components were originally designed based on the 'fail safe' principle. New and more restrictive interpretation of this requirement may require design changes at the system and component level which is impracticable. Note that assessment of all special safety system failures is demonstrated in the safety analysis and already covered in AI 090739 under GIO-009 Update safety analysis to align with REGDOC-2.4.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.11-13_5.2.2.10_15
Document ID	CSA N290.11-13
Article/Clause	5.2.2.10
Requirement Assessed	Assessment of the consequences of the delay or error during the execution of manual actions.
Macro-Gap	SF01-03-15
Issue/Gap Description	Events initiated as a result of human errors in operation and maintenance are not explicitly identified
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce A Safety Report. Bruce A and Bruce B designs incorporate engineered barriers and features to prevent failures from errors in operation and maintenance that could result in harmful consequences in terms of redundancy, diversity and separation. However, these provisions were not always explicitly defined as requirements in terms of errors in operation and maintenance. As a general requirement it is not possible to make additional wholesale design changes to prevent the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences. In addition, Operating Policies and Principles and supporting operating documentation and operating and maintenance procedures provide additional barriers to minimize the likelihood of events initiated as a result of operator errors or errors in maintenance. Capability of the current design of SSCs resulting from human errors will be analyzed as part of AI 090739 under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. Errors in maintenance are also considered as leading to equipment failures which are covered under PIEs and the event classification. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.11-13_5.2.2.10_16
Document ID	CSA N290.11-13
Article/Clause	5.2.2.10
Requirement Assessed	Assessment of the consequences of the delay or error during the execution of manual actions.
Macro-Gap	SF01-03-16
Issue/Gap Description	Clause 5.2.2.10 requires an assessment of the consequences of the delay or error during the execution of manual actions required to recall a heat sink to be completed with respect to meeting the success criteria defined in Clause 4.2. The list of internal initiating events is presented in Table 2-1 (Shutdown Cooling and Maintenance Cooling System Failures) of Part 3 of the Safety Report [8]; however events initiated as a result of human errors in operation and maintenance are not explicitly identified, although initiating event frequencies implicitly include any relevant operator error that may cause the initiating event.
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce B Safety Report. Bruce A and Bruce B design incorporate engineered barriers and features to prevent failures from errors in operation and maintenance that could result in harmful consequences in terms of redundancy, diversity and separation. However, these provisions were not always explicitly defined as requirements in terms of errors in operation and maintenance. As a general requirement it is not possible to make additional wholesale design changes to prevent the possibility of failure of engineered barriers from errors in operation and maintenance that could result in harmful consequences. In addition, Operating Policies and Principles and supporting operating documentation and operating and maintenance procedures provide additional barriers to minimize the likelihood of events initiated as a result of operator errors or errors in maintenance. Capability of the current design of SSCs resulting from human errors will be analyzed as part of AI 090739 under GIO-009 Update Safety Analysis to Align with REGDOC-2.4.1. Errors in maintenance are also considered as leading to equipment failures which are covered under PIEs and the event classification. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_10.1_15
Document ID	CSA N290.3-11
Article/Clause	10.1
Requirement Assessed	Addresses the instrumentation and monitoring requirements.
Macro-Gap	SF01-03-15
Issue/Gap Description	The design documentation does not reflect the requirement that the effect of atmospheric pressure fluctuations due to extreme weather (e.g. tornados) is to be considered in the design of instrumentation.
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce A Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant as it impacts the fundamental design of all containment system instrumentation which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_10.1_16
Document ID	CSA N290.3-11
Article/Clause	10.1
Requirement Assessed	Clause 10.1 requires the effect of atmospheric pressure fluctuations due to extreme weather (e.g., tornados) to be considered in the design of instrumentation
Macro-Gap	SF01-03-16
Issue/Gap Description	The design documentation does not show that the effect of atmospheric pressure fluctuations due to extreme weather (e.g., tornados) was considered in the design of instrumentation.
Rationale	Bruce Power is in compliance with the current licensing basis as documented in the Bruce B Safety Report. Such a requirement is considered to be specifically applicable to a new plant as it impacts the fundamental design of all containment system instrumentation which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_14.1_15
Document ID	CSA N290.3-11
Article/Clause	14.1
Requirement Assessed	Describes the requirements applicable to maintenance of isolation barriers.
Macro-Gap	SF01-05-15
Issue/Gap Description	When maintenance is performed on a penetration, it's required that a single closed isolation barrier shall be demonstrated to be available. The requirement that this barrier should not rely on air or power to maintain its position is not considered in the design documentation.
Rationale	Such a requirement is considered to be specifically applicable to a new plant. Provision of manual isolation would require redesign of all affected isolation barriers and associated systems and components which is impracticable. Operating and maintenance procedures include provisions to ensure appropriate barriers are in place to assure safety when maintenance is performed. Note that current safety analysis demonstrates compliance with the relevant dose acceptance criteria based on the current means of isolation in the as built plant. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_5.5_5.7_15
Document ID	CSA N290.3-11
Article/Clause	5.5_5.7
Requirement Assessed	Section 5.5: For each plant (existing and new build), the scope of DBAs and BDBAs shall be as agreed upon by the authority having jurisdiction (AHJ) and the licensee. Note: The events (DBAs, BDBAs) for new builds and existing plants might differ. Section 5.7: Following a BDBA, the structural integrity of the containment boundary shall be maintained to the extent agreed upon by the AHJ and the licensee.
Macro-Gap	SF01-11-15
Issue/Gap Description	The consequences of severe accidents are mitigated by SAMG; however the current design documentation does not explicitly consider the load conditions during severe accidents.
Rationale	Bruce A and Bruce B accident analyses address dual failures which include a set of DBAs as well as BDBAs as would be classified using modern standards. As such Bruce A and B meet the current licensing basis which considers loading for Design Basis Accidents (DBAs) and some Beyond Design Basis Accidents (BDBAs). Such a requirement is considered to be specifically applicable to a new nuclear plant. This would require a fundamentally different approach in the design of SSCs and implementation of changes for the prevention and mitigation of radiation hazards associated with severe accidents which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_A.2.3_16
Document ID	CSA N290.3-11
Article/Clause	A.2.3
Requirement Assessed	Clause A.2.3 requires pipes open to containment less than 1 hour per year to have two means of isolation (i.e., one of two normally closed manual isolation barriers in series, or two automatic isolation valves, or a combination of a manual and an automatic barrier
Macro-Gap	SF01-19-16
Issue/Gap Description	Single means of isolation for containment
Rationale	Bruce B is in compliance with the current licensing basis which does not include this requirement. Such a requirement is considered to be specifically applicable to a new nuclear plant. Provision of double isolation would require redesign of all affected extensions of the containment boundary associated systems and components which is impracticable. Note that current safety analysis demonstrates compliance with the relevant dose acceptance criteria based on the current means of isolation in the as built plant. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_A.2.5_16
Document ID	CSA N290.3-11
Article/Clause	A.2.5
Requirement Assessed	Clause A.2.5 provides the conditions for having at least one barrier, one condition being pipes with less than 50 mm nominal diameter.
Macro-Gap	SF01-19-16
Issue/Gap Description	N290.3 Clause A.2.5 provides the conditions for having at least one barrier, one condition being pipes with less than 50 mm nominal diameter. Bruce B DG [11] specifies these conditions for pipes with less than 1 inch nominal diameter.
Rationale	Bruce B is in compliance with the current licensing basis which does not include this requirement. Installation of any additional barrier on piping less than 50mm nominal diameter would require a design modification which would also impact the associated piping, supports, containment penetration, concrete and embedded parts which is impracticable. The incremental safety benefit would be marginal as the current design and plant configuration analyzed in the Safety Report complies with the current licencing basis and associated dose acceptance criteria. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_A.3.1_16
Document ID	CSA N290.3-11
Article/Clause	A.3.1
Requirement Assessed	Clause A.3.1 requires that, for pipes connected to HTS (reactor coolant system) with nominal diameter greater than 25 mm, two isolation barriers be provided, one inside and one outside the containment.
Macro-Gap	SF01-19-16
Issue/Gap Description	N290.3 Clause A.3.1 requires that, for pipes connected to HTS (reactor coolant system) with nominal diameter greater than 25 mm, two isolation barriers be provided, one inside and one outside the containment. Bruce B DG [11] (Section 6.2.2.3.1) specifies the requirement for pipes with nominal diameter greater than 1 inch. It also allows both valves to be on only one side of the containment in certain circumstances.
Rationale	Bruce B is in compliance with the current licensing basis which does not include this requirement. Installation of two isolation barriers , one inside and one outside the containment to piping connected to HTS (reactor coolant system) with nominal diameter greater than 25 mm would require a design modification which would also impact the associated piping, supports, containment penetration, concrete and embedded parts which is impracticable. The incremental safety benefit would be marginal as the current design and plant configuration analyzed in the Safety Report complies with the current licencing basis and associated dose acceptance criteria. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF01_CSA N290.3-11_A.3.4_16
Document ID	CSA N290.3-11
Article/Clause	A.3.4
Requirement Assessed	Clause A.3.4 requires that, for pipes connected to HTS (reactor coolant system) with nominal diameter less than 25 mm, at least one barrier be provided
Macro-Gap	SF01-19-16
Issue/Gap Description	N290.3 Clause A.3.1 requires that, for pipes connected to HTS (reactor coolant system) with nominal diameter greater than 25 mm, two isolation barriers be provided, one inside and one outside the containment. Bruce B DG [11] (Section 6.2.2.3.1) specifies the requirement for pipes with nominal diameter greater than 1 inch. It also allows both valves to be on only one side of the containment in certain circumstances
Rationale	Bruce B is in compliance with the current licensing basis which does not include this requirement. Installation of two isolation barriers , one inside and one outside the containment to piping connected to HTS (reactor coolant system) with nominal diameter greater than 25 mm would require a design modification which would also impact the associated piping, supports, containment penetration, concrete and embedded parts which is impracticable. The incremental safety benefit would be marginal as the current design and plant configuration analyzed in the Safety Report complies with the current licencing basis and associated dose acceptance criteria. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.2 Safety goals
Requirement Assessed	Qualitative safety goals A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. Quantitative application of the safety goals For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency 2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. Core damage frequency The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-5 per reactor year. Small release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E15 becquerels of iodine-131 shall be less than 1E-5 per reactor year. A greater release may require temporary

	evacuation of the local population. Large release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E14 becquerels of cesium-137 shall be less than 1E-6 per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.
Macro-Gap	SF05-06-15
Issue/Gap Description	Bruce A safety goals are less restrictive (larger) than those proposed for new plants (Gap 1). See SFR 6 for more details.
Rationale	Bruce A is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement. The more restrictive requirements are considered to be specifically applicable to a new nuclear plant as such goals dictate the technical basis

	of NPP design starting from siting to decommissioning. Originally, Bruce A and Bruce B SSCs were not designed to meet the more stringent Safety Goal Limits as prescribed for new NPP design. Such limits affect the plant design as a whole and would have required a fundamentally different approach in the design of SSCs and various options to be pursued to meet the more stringent safety goals. It is judged that it is impracticable to implement such a change at the plant level. However, Bruce Power continues to make practicable design changes to improve plant safety and safety analysis margins including the current safety goals. For example, as demonstrated in SFR-6, PSA results based on the improvements made as a result of follow-up actions and initiatives to the Fukushima event for both Bruce A and Bruce B. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.2 Safety goals
Requirement Assessed	Qualitative safety goals A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. Quantitative application of the safety goals For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency 2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. Core damage frequency The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-5 per reactor year. Small release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E15 becquerels of iodine-131 shall be less than 1E-5 per reactor year. A greater release may require temporary

	evacuation of the local population. Large release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E14 becquerels of cesium-137 shall be less than 1E-6 per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.
Macro-Gap	SF05-10-16
Issue/Gap Description	The Bruce B safety goals are less restrictive (larger) than those proposed for new plants.
Rationale	Bruce B is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement. The more restrictive requirements are considered to be specifically applicable to a new nuclear plant as such goals dictate the technical basis

	of NPP design starting from siting to decommissioning. Originally, Bruce A and Bruce B SSCs were not designed to meet the more stringent Safety Goal Limits as prescribed for new NPP design. Such limits affect the plant design as a whole and would have required a fundamentally different approach in the design of SSCs and various options to be pursued to meet the more stringent safety goals. It is judged that it is impracticable to implement such a change at the plant level. However, Bruce Power continues to make practicable design changes to improve plant safety and safety analysis margins including the current safety goals. For example, as demonstrated in SFR-6, PSA results based on the improvements made as a result of follow-up actions and initiatives to the Fukushima event for both Bruce A and Bruce B. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF05-02-16
Issue/Gap Description	DECs were not considered in the design basis.
Rationale	Bruce B plant design basis, documented in the Bruce B Safety Report, demonstrates that plant SSCs will perform as designed to meet regulatory dose limits as stipulated in the PROL. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant. Design Extension Condition (DEC) is a plant state introduced as a subset of BDBAs that have to be considered for the prevention and mitigation of radiation exposures in the plant design. Design requirements associated with such a change affects the plant design as a whole and requires definition of a set of new design limits and acceptance criteria to be established as a pre-requisite. This would require a fundamentally different approach in the design of SSCs and

	implementation of changes for the prevention and mitigation of radiation hazards associated with DBAs, DECs and BDBAs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_6.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF05-02-15
Issue/Gap Description	The limits for AOOs are currently taken to be the same as for DBAs (Gap 3).

Rationale	Such a requirement is considered to be specifically applicable to a new plant. Dose limits for AOOs for new plants are more conservative (an order of magnitude) than those limits applicable to single failures limits used in the original design. This would require a fundamentally different approach in the design of affected SSCs and implementation of changes for the prevention and mitigation of radiation hazards associated with AOOs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_6.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF05-02-16
Issue/Gap Description	The limits for AOOs are currently taken to be the same as for DBAs.

Rationale	Such a requirement is considered to be specifically applicable to a new plant. Dose limits for AOOs for new plants are more conservative (an order of magnitude) than those limits applicable to single failures limits used in the original design. This would require a fundamentally different approach in the design of affected SSCs and implementation of changes for the prevention and mitigation of radiation hazards associated with AOOs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_6.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF05-02-16
Issue/Gap Description	Provisions for DECs were not explicitly considered in the design basis, and therefore Bruce B does not meet this requirement which is intended for

	new builds.
Rationale	Bruce Power is in compliance with the current licensing basis which is documented in the Bruce B Safety Report. Although DECs were not considered explicitly in the original design basis, the current safety analysis includes some event sequences that would be categorized as BDBAs which demonstrate the as built capabilities of the plant SSCs against such events. Bruce Power has also implemented practicable design changes to improve mitigation of BDBAs/DECs such as installation of PARs, provision of alternate heat sinks in response to Fukushima and other international OPEX. Such a requirement is considered to be specifically applicable to a new nuclear plant. Design Extension Condition (DEC) is a plant state introduced as a subset of BDBAs that have to be considered for the prevention and mitigation of radiation exposures in the plant design. Design requirements associated with such a change affects the plant design as a whole and requires definition of a set of new design limits and acceptance criteria to be established as a pre-requisite. This would require a fundamentally different approach in the design of SSCs and implementation of changes for the prevention and mitigation of radiation hazards associated with DBAs, DECs and BDBAs which is impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_8.10.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.10.4 Credit for operator action
Requirement Assessed	If operator action is required for actuation of any safety system or safety support system equipment, all of the following requirements shall apply: 1. there are clear, well-defined, validated, and readily available operating procedures that identify the necessary actions 2. there is instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action 3. following indication of the necessity for operator action inside the control rooms, there are at least 30 minutes available before the operator action is required 4. following indication of the necessity for operator action outside the control rooms, there is a minimum of 1 hour available before the operator action is required For automatically initiated safety systems and control logic actions, the design shall facilitate backup manual initiation from inside the appropriate control room. Guidance The design should ensure that no failure of monitoring or display systems will influence the functioning of other safety systems. The available time before operator action can be credited should be counted from the receipt of an unambiguous indication of a potential accident (typically an alarm) and includes diagnostic time. The time available to perform the actions should be based on the analysis of the plant response to AOOs and DBAs, using realistic assumptions. The time required for operator action should be based on a human factors engineering analysis of operator response time, which (in turn) is based on a documented sequence of operator actions. Uncertainties in the analysis of time required are identified and assessed. An adequate time margin should also be added to the analyzed time. If operator action is required for actuation of any safety function, other than meeting the requirements of this regulatory document, the analysis should also demonstrate that: • there is sufficient time available for the operator to perform the required manual action

	• the operator can perform the actions correctly and reliably in the time available The sequence of actions should use only alarms, controls, and displays that would be available in locations where the tasks will be performed and should be available in all scenarios analysed. A preliminary validation should be conducted, to provide independent confirmation to the validity of the estimated “time available” and “time required” for human actions. The preliminary validation results should support the conclusion that the time required, including margin, to perform individual steps and the overall documented sequence of manual operator actions are reasonable, realistic, repeatable, and bounded by the initial analysis. An integrated system test should also be conducted, to validate the manual actions credited in the safety analysis, using a full-scale simulator. Tasks conducted outside the control room should be included in the integrated system validations. Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field location.
Macro-Gap	SF05-02-15
Issue/Gap Description	Operator actions in Part 3 of the Safety Report are assumed to be 15 minutes for actions inside the control room and 30 minutes for actions outside the control room. These assumptions do not meet the proposed values of REGDOC-2.5.2 for new plants (Gap 1). They are consistent with the guidance of REGDOC-2.4.1 and CSA 290.1.
Rationale	Bruce Power is in compliance with the current licensing basis which puts operator action times inside the control room at 15 minutes as documented in the Bruce A Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. Changing operator action time from 15 minutes to 30 minutes for actions inside the control room and from 30 minutes to 1 hour for actions outside will require fundamental changes to safety and safety support system design. For example, this would require changes to meet new requirements in terms of SSC actuation, capacity, and performance, etc., which is impracticable. Also as noted in the gap description assumptions used in the safety analysis are consistent with the guidance of REGDOC 2.4.1 and CSA 290.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF05_CNSC REGDOC 2.5.2_8.10.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.10.4 Credit for operator action
Requirement Assessed	If operator action is required for actuation of any safety system or safety support system equipment, all of the following requirements shall apply: 1. there are clear, well-defined, validated, and readily available operating procedures that identify the necessary actions 2. there is instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action 3. following indication of the necessity for operator action inside the control rooms, there are at least 30 minutes available before the operator action is required 4. following indication of the necessity for operator action outside the control rooms, there is a minimum of 1 hour available before the operator action is required For automatically initiated safety systems and control logic actions, the design shall facilitate backup manual initiation from inside the appropriate control room. Guidance The design should ensure that no failure of monitoring or display systems will influence the functioning of other safety systems. The available time before operator action can be credited should be counted from the receipt of an unambiguous indication of a potential accident (typically an alarm) and includes diagnostic time. The time available to perform the actions should be based on the analysis of the plant response to AOOs and DBAs, using realistic assumptions. The time required for operator action should be based on a human factors engineering analysis of operator response time, which (in turn) is based on a documented sequence of operator actions. Uncertainties in the analysis of time required are identified and assessed. An adequate time margin should also be added to the analyzed time. If operator action is required for actuation of any safety function, other than meeting the requirements of this regulatory document, the analysis should also demonstrate that: • there is sufficient time available for the operator to perform the required manual action

	• the operator can perform the actions correctly and reliably in the time available The sequence of actions should use only alarms, controls, and displays that would be available in locations where the tasks will be performed and should be available in all scenarios analysed. A preliminary validation should be conducted, to provide independent confirmation to the validity of the estimated “time available” and “time required” for human actions. The preliminary validation results should support the conclusion that the time required, including margin, to perform individual steps and the overall documented sequence of manual operator actions are reasonable, realistic, repeatable, and bounded by the initial analysis. An integrated system test should also be conducted, to validate the manual actions credited in the safety analysis, using a full-scale simulator. Tasks conducted outside the control room should be included in the integrated system validations. Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field location.
Macro-Gap	SF05-08-16
Issue/Gap Description	Operator actions in Part 3 of the Safety Report are assumed to be 15 minutes for actions inside the control room and 30 minutes for actions outside the control room. These assumptions do not meet the proposed values of REGDOC-2.5.2 for new plants.
Rationale	Bruce Power is in compliance with the current licensing basis which puts operator action times inside the control room at 15 minutes as documented in the Bruce B Safety Report. Such a requirement is considered to be specifically applicable to a new nuclear plant. Changing operator action time from 15 minutes to 30 minutes for actions inside the control room and from 30 minutes to 1 hour for actions outside will require fundamental changes to safety and safety support system design. For example, this would require changes to meet new requirements in terms of SSC actuation, capacity, and performance, etc., which is impracticable. Also as noted elsewhere assumptions used in the safety analysis are consistent with the guidance of REGDOC 2.4.1 and CSA 290.1. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF06_CNSC REGDOC 2.5.2_4.2.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.2 Safety goals
Requirement Assessed	Qualitative safety goals A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. Quantitative application of the safety goals For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency 2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. Core damage frequency The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-5 per reactor year. Small release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E15 becquerels of iodine-131 shall be less than 1E-5 per reactor year. A greater release may require temporary

	evacuation of the local population. Large release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E14 becquerels of cesium-137 shall be less than 1E-6 per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.
Macro-Gap	SF06-01-15
Issue/Gap Description	- Although the result of each separate PRA meets the safety goal limits set up for Bruce A PRAs, their aggregates obtained by respective summation of SCDFs, SRFs and LRFs and across all available PRA types, do not meet the more stringent quantitative safety goal targets set up in the requirement clause. Therefore, a gap is assessed against this clause.
Rationale	Bruce A is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement.

	The more restrictive requirements are considered to be specifically applicable to a new nuclear plant as such goals dictate the technical basis of NPP design starting from siting to decommissioning. Originally, Bruce A and Bruce B SSCs were not designed to meet the more stringent Safety Goal Limits as prescribed for new NPP design. Such limits affect the plant design as a whole and would have required a fundamentally different approach in the design of SSCs and various options to be pursued to meet the more stringent safety goals. It is judged that it is impracticable to implement such a change at the plant level. However, Bruce Power continues to make practicable design changes to improve plant safety and safety analysis margins including the current safety goals. For example, as demonstrated in SFR-6, PSA results based on the improvements made as a result of follow-up actions and initiatives to the Fukushima event for both Bruce A and Bruce B. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF06_CNSC REGDOC 2.5.2_4.2.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.2 Safety goals
Requirement Assessed	Qualitative safety goals A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. Quantitative application of the safety goals For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency 2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. Core damage frequency The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 1E-5 per reactor year. Small release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E15 becquerels of iodine-131 shall be less than 1E-5 per reactor year. A greater release may require temporary

	evacuation of the local population. Large release frequency The sum of frequencies of all event sequences that can lead to a release to the environment of more than 1E14 becquerels of cesium-137 shall be less than 1E-6 per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.
Macro-Gap	SF06-01-16
Issue/Gap Description	The Bruce B PRA results to date indicate that the sum of frequencies of all event sequences that lead to severe core damage is greater than 1E-5 per reactor year, and that the sum of frequencies of all event sequences that lead to large release is greater than 1E-6 per reactor year.
Rationale	Bruce B is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement.

	The more restrictive requirements are considered to be specifically applicable to a new nuclear plant as such goals dictate the technical basis of NPP design starting from siting to decommissioning. Originally, Bruce A and Bruce B SSCs were not designed to meet the more stringent Safety Goal Limits as prescribed for new NPP design. Such limits affect the plant design as a whole and would have required a fundamentally different approach in the design of SSCs and various options to be pursued to meet the more stringent safety goals. It is judged that it is impracticable to implement such a change at the plant level. However, Bruce Power continues to make practicable design changes to improve plant safety and safety analysis margins including the current safety goals. For example, as demonstrated in SFR-6, PSA results based on the improvements made as a result of follow-up actions and initiatives to the Fukushima event for both Bruce A and Bruce B. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF06_CNSC REGDOC 2.5.2_8.4.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.4.2 Reliability
Requirement Assessed	The design shall permit ongoing demonstration that each means of shutdown is being operated and maintained in a manner that ensures continued adherence to reliability and effectiveness requirements. Periodic testing of the systems and their components shall be scheduled at a frequency commensurate with applicable requirements. Guidance The reliability calculation should include sensing the need for shutdown, initiation of shutdown, and insertion of negative reactivity. All elements necessary to complete the shutdown function should be included. The reliability of the shutdown function should be such that the cumulative frequency of failure to shutdown on demand is less than 1E-5 failures per demand, and the contribution of all sequences involving failure to shutdown to the large release frequency of the safety goals is less than 1E-7/yr. This considers the likelihood of the initiating event and recognizes that the two shutdown means may not be completely independent. Section 7.6.2 requires that the shutdown function be delivered even in the presence of any single failure and even during the worst configuration from testing and maintenance. For example, for a rod based system to meet the SFC, the safety analysis may assume that the two highest worth control rods are unavailable (one for testing, and one assumed to fail on demand, in accordance with the SFC). In this case, no further testing of rods would be allowed until the rod under testing becomes available.
Macro-Gap	SF06-02-15
Issue/Gap Description	The proposed safety goal that the contribution to the large release frequency from all sequences involving failure to shutdown be below 10-7/yr events per reactor per year is not met.
Rationale	Bruce A is in compliance with the current licensing basis which aligns with the SCDF and LRF goals which were accepted into the licensing basis when PSA was adopted as a licensing requirement. Such a requirement is considered to be specifically applicable to a new plant. Reducing the frequency of failure to shut down below 1E-5 to meet the LRF limit of 1E-7 would require fundamental design changes to the shutdown systems which are impracticable. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits.

Gap #	SF07_CNSC REGDOC 2.5.2_7.13.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.13.1 Seismic design and classification
Requirement Assessed	The design authority shall ensure that seismically qualified SSCs important to safety are qualified to a design-basis earthquake (DBE), and ensure that they are categorized accordingly. This shall apply to: 1. SSCs whose failure could directly or indirectly cause an accident leading to core damage 2. SSCs restricting the release of radioactive material to the environment 3. SSCs that assure the subcriticality of stored nuclear material 4. SSCs such as radioactive waste tanks containing radioactive material that, if released, would exceed regulatory dose limits The design of these SSCs shall also meet the DBE criteria to maintain all essential attributes, such as pressure boundary integrity, leak-tightness, operability, and proper position in the event of a DBE. The design shall ensure that no substantive damage to these SSCs will be caused by the failure of any other SSC under DBE conditions. Seismic fragility levels shall be evaluated for SSCs important to safety by analysis or, where possible, by testing. A beyond-design-basis earthquake (BDBE) shall be identified that meets the requirements for identification of DECs as described in section 7.3.4. SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions. Such demonstration shall provide high confidence of low probability of failure (HCLPF) under BDBE conditions for these SSCs. This demonstration need not be seismic qualification by testing. Guidance The seismic design of an NPP should account for: • technical safety objectives and corresponding load categories • seismic input motion • seismic classification • structural layout criteria • seismic analysis and design of structural systems, subsystems and equipment • seismic testing and instrumentation

	Design and beyond design load categories are defined to demonstrate structural performance in operational states, DBAs and DECs. In addition, beyond design load categories are considered for structural performance in DECs. Earthquake load is not part of the normal load category corresponding to normal operation. Site design earthquake load, according to the CSA N289 series on seismic design and qualification, is defined under the severe load category corresponding to AOO. A DBE is defined as a part of the abnormal or extreme load category corresponding to DBA. BDBE load should be considered under DECs. Seismic input motion, derived from the DBE, should be based on seismicity and geologic conditions at the site and expressed in such a manner that it can be applied for the qualification of SSCs. The DBE is defined by multiplying the mean site specific uniform hazard spectrum with a probability of occurrence of 1E-4/yr by a design factor, defined in the standard ASCE 43-05, Seismic Design Criteria for Structures, Systems and Components in Nuclear Facilities. The probability of occurrence of the defined DBE is therefore equivalent to the probability of DBAs. A minimum seismic input motion, consistent with national or international standards, should be considered in the design phase for the DBE. The minimum seismic input motion should take into account frequencies of interest for SSCs. Structural layout criteria, including structural separation, should follow best engineering practices and lessons learned from past earthquakes. Modelling of soil-structure interaction (SSI) should be based on geotechnical investigation and taking into account the random nature of soil material properties and inherent uncertainties incorporated in soil constitutive models used in the analysis. To account for uncertainties in soil properties a range with at least three values (upper limit, best estimate and lower limit) should be taken into account in the analysis according to CSA N289.3, Design procedures for seismic qualification of nuclear power plants, clause 5.2.3. The analysis of SSI should take into account all effects due to kinematic interaction (effect of applied seismic ground motion on massless structure) and inertial interaction (inertial forces developed in the structure due to the seismic ground motion). The detail and sophistication of soil-structure models should be in accordance with the purposes of the analyses. The frequency range of interest determines aspects of the structure model and the SSI model parameters. The frequency range of interest should be based on the combination of the frequency range of the earthquake input, the soil properties, the frequency range of building response (including response of subsystems modelled in the main building or structure model), and the frequency range of the response parameter of interest. Refined finite element meshes and increased analytical rigor are required to transmit higher frequencies through the analytical models.

	Damping ratios for structural systems and sub-systems should be taken into account according to recognized standards such as ASCE 43-05 and CSA N289.3. For generating the in-structure response spectra to be used as input to the structure mounted systems and components, Response Level 1 damping of the structure is more appropriate unless the structure response generally exceeds demand over capacity factor given in ASCE 43-05. The seismic design of structural systems should be categorized according to seismic design category (SDC) 1 to 5 as per ASCE 43-05. SDC 1 and 2 structural systems should be in accordance with the National Building Code of Canada, Division B, Part 4. According to the Code, SDC 1 should be as normal and SDC 2 as post-disaster. All structures important to safety are classified as SDC 5. However, the designer may still classify some structures as SDC 3, 4 and 5 provided that they include proper justification. Guidance on SDC 3, 4 and 5 (if SDC 3 and 4 are used) structural systems are provided as follows: • for concrete containment, the design should be based on the American Society of Civil Engineers, ASCE 43-05 (SDC 5, limit state D) and CSA N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants • for steel containment, the design should be based on ASCE 43-05 (SDC 5), 2010 ASME Boiler and Pressure Vessel Code, Section III: Rules for Construction of Nuclear Power Plant Components, Division 1, Subsection NE: Class MC Components and U.S. NRC Regulatory Guide 1.57, Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components • for concrete and steel safety related structures the design should be based on ASCE 43-05 (SDC 5, limit state D) and CSA N291, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants For all safety design categories in an NPP, ductility requirements should be in accordance with CSA-A23.3, Design of Concrete Structures for concrete structures and CSA S16, Design of Steel Structures for steel structures assuming that the structures are ductile or type D. These ductility requirements should provide margins for the BDBE. Sub-system analysis should follow the guidance presented for structural systems with the following criteria specific to sub-system supports: • in-structure response spectra • in-structure time response histories The methods of defining in-structure response spectra or in-structure time-histories as well as application of this seismic input to sub-systems and

	components should be in accordance with ASCE 04, Seismic Analysis for Safety-Related Nuclear Structures. Multiple support seismic input of sub-systems and components should take into account their inertial and kinematic components. The analysis should follow ASCE 04 or CSA N289.3, Design procedures for seismic qualification of nuclear power plants. Determination of the number of earthquake cycles for sub-system analysis should be in accordance with U.S. NRC NUREG-0800, Standard Review Plan, section 3.7.3, Seismic Subsystem Analysis as well as seismic analysis of above-ground tanks. Seismic design of sub-systems and components should be in accordance with ASCE 43-05 section 8.2.3 which follows ASME Code. For equipment qualified by testing, multi-axis, multi-frequency testing is acceptable for the DBE in accordance with the requirement of IEEE 344-2004 – IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations and that the testing response spectrum should be at least a factor of 1.4 times the required response spectrum throughout the frequency range. Any deviation from this should be conservatively justified on a case-by-case basis. Any evaluation for BDBE should utilize the methodology in the Electrical Power Research Institute, (EPRI) TR-103959, Methodology for Developing Seismic Fragilities to determine if a HCLPF goal is met. Seismic instrumentation design should follow CSA-N289.5, Seismic Instrumentation Requirements for Nuclear Power Plants and Nuclear Facilities which itemizes the requirements for single and multiple unit site seismic instrumentation. Beyond-design-basis margin should be such that seismically induced SSC failure probabilities do not contribute to the total core damage frequency and small and large release frequency to the extent that they do not meet the safety goals. To support meeting the safety goals, the acceptance criterion for BDBE should demonstrate that the plant HCLPF is at least 1.67 times the DBE. Assessment and validation of margins for beyond-design-basis earthquakes should be considered, including the metric HCLPF. The seismic isolation of SSCs is an acceptable design approach to limit seismic demand. Seismic isolation devices should be designed, manufactured and installed to withstand a seismic action defined by a DBE without any failure, preserving its mechanical resistance and full load bearing capacity during and after the earthquake. Moreover, the devices and the whole structural system should be designed to withstand a BDBE up to 2 times the spectral accelerations of the DBE without major damage

	and preserving its function. It includes the provisions to accommodate the structural displacements up to 2 times the displacements under DBE conditions.
Macro-Gap	SF07-01-16
Issue/Gap Description	CNSC REGDOC 2.5.2 clause 7.13.1 invokes the CSA N289 series for detailed guidance, including definition of a DBE. In CSA N289.1-08 (R013), a DBE is defined as having a probability of exceedance of 1E-4/a, or such level as determined by the regulatory authority; furthermore, the definition notes that the DBE for some older plants was based on a probability of exceedance of 1E-3/a. While the ground response spectra used in the seismic qualification of Bruce B are based on a probability of exceedance of 1E-3/a, the wording of the DBE definition in the standard, including the note about older plants, implicitly permits this. This is considered a Gap against the guidance of CSA N289.1-08 regarding DBE definition.
Rationale	The design basis earthquake for Bruce B is peak ground acceleration of 0.05g was selected to correspond to an occurrence rate of less than 1E-3 per year. It is not practicable to make wholesale design changes to comply with the intent of this clause which impacts all SSCs important to safety that drives all structural and component level design requirements that is applicable to new plants as it can only be implemented at the initial design stage. It should also be noted that Bruce Power has received a formal interpretation of Clause 4.2 of CSA Standard N289.3-10 which states that the intent of the clause is applicable only to the design of SSCs of new nuclear power plants [NK29-CORR-00531-12453]. This micro-gap has been judged to have low safety significance, and would have high resource usage to realize the marginal benefits

Appendix E – CATEGORY 2: Safety Improvement Considered Unnecessary to Implement as Part of IIP

Appendix E consists of those micro-gaps identified in the Safety Factor Reports for which safety improvements are considered unnecessary to implement as part of the IIP.

- Table 49 provides a consolidation of all micro-gaps within this category. It is ordered such that gaps that are similar or identical appear consecutively. This can be regarded as a "smart table of contents" for the micro-gaps discussed in the next bullet, and provides a direct linkage back to the origin of the micro-gaps in the Safety Factor Reports.
- Table 50 provides the details for each of the micro-gaps within this category. This is based on an export from the PSR database, and is ordered first by Safety Factor, then by regulatory document/code/standard, then by clause.
- Table 51 provides the details for each of the micro-gaps that were identified by the CNSC review of the SFRs and the GAR/IIP (Revision R01).

The micro-gap number, which is provided in both tables, facilitates their use.

Table 49: Consolidation of Micro-gaps Considered Unnecessary to Implement as Part of IIP

Category 2- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_ASME B31.1_MISC_15	SF01-17-15	ASME B31.1	MISC	2
SF01_ASME BPVC Section III_MISC_16	SF01-17-16	ASME BPVC Section III	MISC	3
SF01_ASME BPVC Section III_MISC_15	SF01-15-15	ASME BPVC Section III	MISC	3
SF01_ASME BPVC Section VIII_MISC_16	SF01-18-16	ASME BPVC Section VIII	MISC	4
SF01_ASME BPVC Section VIII_MISC_15	SF01-16-15	ASME BPVC Section VIII	MISC	4
SF01_CNSC REGDOC 2.5.2_7.6.3_15	SF01-01-15	CNSC REGDOC 2.5.2	7.6.3	20
SF01_CNSC REGDOC 2.5.2_7.6.3_16	SF01-20-16	CNSC REGDOC 2.5.2	7.6.3	20
SF01_CNSC REGDOC 2.5.2_7.11_16	SF01-08-16	CNSC REGDOC 2.5.2	7.11	24
SF01_CNSC REGDOC 2.5.2_7.11_15	SF01-08-15	CNSC REGDOC 2.5.2	7.11	24
SF01_CNSC REGDOC 2.5.2_7.13_16	SF01-15-16	CNSC REGDOC 2.5.2	7.13	25
SF01_CSA N289.1_6.5.6.3_16	SF01-15-16	CSA N289.1	6.5.6.3	25
SF01_SF1 RT_5.4_16	SF01-21-16	SF1 RT	5.4	72
SF01_CSA N289.1_6.5.6.4_16	SF01-15-16	CSA N289.1	6.5.6.4	25
SF03_CSA N289.1_6.5.6.4_16	SF03-02-16	CSA N289.1	6.5.6.4	25

Category 2- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF03_CSA N289.1_6.5.6.3_16	SF03-02-16	CSA N289.1	6.5.6.3	25
SF01_CNSC REGDOC 2.5.2_8.9_16	SF01-12-16	CNSC REGDOC 2.5.2	8.9	40
SF01_CNSC REGDOC 2.5.2_8.9_15	SF01-12-15	CNSC REGDOC 2.5.2	8.9	40
SF01_CNSC REGDOC 2.5.2_8.9.2_16	SF01-12-16	CNSC REGDOC 2.5.2	8.9.2	41
SF01_CNSC REGDOC 2.5.2_8.9.2_15	SF01-12-15	CNSC REGDOC 2.5.2	8.9.2	41
SF01_CNSC REGDOC 2.5.2_8.13.3_15	SF01-14-15	CNSC REGDOC 2.5.2	8.13.3	49
SF01_CNSC REGDOC 2.5.2_8.13.3_16	SF01-14-16	CNSC REGDOC 2.5.2	8.13.3	49
SF01_CSA N290.0-11_4.9-4.13_16	SF01-05-16	CSA N290.0-11	4.9-4.13	59
SF01_CSA N290.2-11_5.12.5_16	SF01-14-16	CSA N290.2-11	5.12.5	64
SF01_CSA N290.2-11_5.12.5_15	SF01-14-15	CSA N290.2-11	5.12.5	64
SF03_CSA N289.1_5.3.11_16	SF03-01-16	CSA N289.1	5.3.11	73
SF03_CSA N289.2_4.4.2.2_16	SF03-03-16	CSA N289.2	4.4.2.2	74
SF03_CSA N289.5_4.1.1.3_16	SF03-04-16	CSA N289.5	4.1.1.3	75
SF05_CNSC REGDOC 2.4.1_4.2.1_15	SF05-02-15	CNSC REGDOC 2.4.1	4.2.1	82
SF05_CNSC REGDOC 2.4.1_4.2.1_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.1	82
SF05_CNSC REGDOC 2.4.1_4.2.1_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.1	83
SF05_CNSC REGDOC 2.4.1_4.2.1_15	SF05-02-15	CNSC REGDOC 2.4.1	4.2.1	83
SF05_CNSC REGDOC 2.4.1_4.2.2.5_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.2.5	263
SF05_CNSC REGDOC 2.4.1_4.2.3_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.3	90
SF05_CNSC REGDOC 2.4.1_4.3.2_15	SF05-01-15	CNSC REGDOC 2.4.1	4.3.2	93
SF05_CNSC REGDOC 2.4.1_4.3.2_16	SF05-02-16	CNSC REGDOC 2.4.1	4.3.2	93
SF05_CNSC REGDOC 2.4.1_4.3.2_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.2	94
SF05_CNSC REGDOC 2.4.1_4.3.2_16	SF05-03-16	CNSC REGDOC 2.4.1	4.3.2	94
SF05_CNSC REGDOC 2.4.1_4.3.2_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.2	96
SF05_CNSC REGDOC 2.4.1_4.3.2_16	SF05-06-16	CNSC REGDOC 2.4.1	4.3.2	96
SF05_CNSC REGDOC 2.4.1_4.3.2_16	SF05-02-16	CNSC REGDOC 2.4.1	4.3.2	97
SF05_CNSC REGDOC 2.4.1_4.3.2_15	SF05-01-15	CNSC REGDOC 2.4.1	4.3.2	97
SF05_CNSC REGDOC 2.4.1_4.3.4_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.4	100
SF05_CNSC REGDOC 2.4.1_4.3.4_16	SF05-03-16	CNSC REGDOC 2.4.1	4.3.4	100
SF05_CNSC REGDOC 2.4.1_4.3.4_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.4	101
SF05_CNSC REGDOC 2.4.1_4.3.4_16	SF05-03-16	CNSC REGDOC 2.4.1	4.3.4	101
SF12_SF12 RT_5.1_16	SF12-01-16	SF12 RT	5.1	108
SF05_CNSC REGDOC 2.4.1_4.4.2.9_16	SF05-01-16	CNSC REGDOC 2.4.1	4.4.2.9	109
SF05_CNSC REGDOC 2.4.1_4.4.2.9_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.2.9	109
SF05_CNSC REGDOC 2.4.1_4.4.4.6_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.4.6	120
SF05_CNSC REGDOC 2.4.1_4.4.4.6_16	SF05-01-16	CNSC REGDOC 2.4.1	4.4.4.6	120
SF05_CNSC REGDOC 2.4.1_4.4.4.7_15	SF05-07-15	CNSC REGDOC 2.4.1	4.4.4.7	121
SF05_CNSC REGDOC 2.4.1_4.4.4.7_16	SF05-06-16	CNSC REGDOC 2.4.1	4.4.4.7	121

Category 2- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF05_CNSC REGDOC 2.4.1_4.4.6_16	SF05-07-16	CNSC REGDOC 2.4.1	4.4.6	123
SF05_CNSC REGDOC 2.4.1_4.4.6_15	SF05-09-15	CNSC REGDOC 2.4.1	4.4.6	123
SF08_SF8 RT_4.1_16	SF08-09-16	SF8 RT	4.1	138
SF08_SF8 RT 2015_4.1_15	SF08-10-15	SF8 RT 2015	4.1	138
SF10_SF10 RT 2015_4.1_15	SF10-04-15	SF10 RT 2015	4.1	138
SF08_SF8 RT_4.7_16	SF08-07-16	SF8 RT	4.7	139
SF08_SF8 RT_5.3_16	SF08-02-16	SF8 RT	5.3	140
SF08_SF8 RT_5.3_16	SF08-12-16	SF8 RT	5.3	141
SF08_SF8 RT_5.7_16	SF08-03-16	SF8 RT	5.7	143
SF08_SF8 RT 2015_5.7_15	SF08-04-15	SF8 RT 2015	5.7	143
SF08_SF8 RT_5.13_16	SF08-06-16	SF8 RT	5.13	145
SF08_SF8 RT 2015_5.13_15	SF08-07-15	SF8 RT 2015	5.13	145
SF08_SF8 RT_5.14.2_16	SF08-01-16	SF8 RT	5.14.2	146
SF08_SF8 RT_7.2_16	SF08-10-16	SF8 RT	7.2	146
SF11_SF11 RT_5.4_15	SF11-05-15	SF11 RT	5.4	146
SF11_SF11 RT_5.4_16	SF11-04-16	SF11 RT	5.4	146
SF08_SF8 RT_7.2_16	SF08-08-16	SF8 RT	7.2	147
SF08_SF8 RT_7.2_16	SF08-11-16	SF8 RT	7.2	148
SF08_SF8 RT 2015_5.13_15	SF08-01-15	SF8 RT 2015	5.13	152
SF08_SF8 RT 2015_5.14.2_15	SF08-02-15	SF8 RT 2015	5.14.2	153
SF09_SF9 RT_5.3.1.2_16	SF09-01-16	SF9 RT	5.3.1.2	154
SF10_SF10 RT_5.2.3_16	SF10-01-16	SF10 RT	5.2.3	155
SF10_SF10 RT 2015_5.2.3_15	SF10-02-15	SF10 RT 2015	5.2.3	155
SF10_SF10 RT_5.3.3_16	SF10-02-16	SF10 RT	5.3.3	156
SF10_SF10 RT 2015_5.3.3_15	SF10-03-15	SF10 RT 2015	5.3.3	156
SF10_SF10 RT_5.3.3_16	SF10-03-16	SF10 RT	5.3.3	157
SF10_SF10 RT 2015_5.2.5_15	SF10-01-15	SF10 RT 2015	5.2.5	158
SF11_CSA N292.3-14_8.7_16	SF11-05-16	CSA N292.3-14	8.7	159
SF11_CSA N292.3-14_9.1_16	SF11-01-16	CSA N292.3-14	9.1	160
SF11_CSA N292.3-14_9.1_15	SF11-02-15	CSA N292.3-14	9.1	160
SF11_CSA N292.3-14_9.2.6_16	SF11-02-16	CSA N292.3-14	9.2.6	161
SF11_CSA N292.3-14_9.2.6_15	SF11-03-15	CSA N292.3-14	9.2.6	161
SF11_CSA N292.3-14_11.2.1_16	SF11-03-16	CSA N292.3-14	11.2.1	162
SF11_CSA N292.3-14_11.2.1_15	SF11-04-15	CSA N292.3-14	11.2.1	162
SF11_CSA N292.3-14_11.2.2_16	SF11-03-16	CSA N292.3-14	11.2.2	162
SF11_CSA N292.3-14_11.2.2_15	SF11-04-15	CSA N292.3-14	11.2.2	162
SF11_CSA N292.3-14_11.2.3_16	SF11-03-16	CSA N292.3-14	11.2.3	162
SF11_CSA N292.3-14_11.2.3_15	SF11-04-15	CSA N292.3-14	11.2.3	162

Category 2- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF12_CSA N290.12_5.2.2_16	SF12-05-16	CSA N290.12	5.2.2	168
SF12_CSA N290.12_5.3.2_16	SF12-05-16	CSA N290.12	5.3.2	172
SF12_CSA N290.12_5.5_16	SF12-05-16	CSA N290.12	5.5	176
SF12_NUREG-0700_Part_II_4_15	SF12-02-15	NUREG-0700	Part_II_4	194
SF12_NUREG-0700_Part_II_4_16	SF12-04-16	NUREG-0700	Part_II_4	194
SF12_NUREG-0700_Part_II_5_15	SF12-02-15	NUREG-0700	Part_II_5	195
SF12_NUREG-0700_Part_II_5_16	SF12-04-16	NUREG-0700	Part_II_5	195
SF12_NUREG-0700_Part_II_10_16	SF12-04-16	NUREG-0700	Part_II_10	198
SF12_NUREG-0700_Part_II_10_15	SF12-02-15	NUREG-0700	Part_II_10	198
SF13_CNSC REGDOC 2.3.2_4.3_15	SF13-03-15	CNSC REGDOC 2.3.2	4.3	215
SF13_SF13 RT 2016_5.1_16	SF13-01-16	SF13 RT 2016	5.1	220
SF15_SF15 RT_5.1.3.3_15	SF15-05-15	SF15 RT	5.1.3.3	238
SF15_WANO GL 2004-01-R1_IV.C2._15	SF15-02-15	WANO GL 2004-01-R1	IV.C2.	238
SF15_SF15 RT_5.6.1_15	SF15-05-15	SF15 RT	5.6.1	241
SF15_WANO GL 2004-01-R1_I.C4._15	SF15-05-15	WANO GL 2004-01-R1	I.C4.	241
SF15_WANO GL 2004-01-R1_I.C2._15	SF15-05-15	WANO GL 2004-01-R1	I.C2.	242
SF15_WANO GL 2004-01-R1_I.C5._15	SF15-01-15	WANO GL 2004-01-R1	I.C5.	243
SF15_WANO GL 2004-01-R1_V.C1._15	SF15-01-15	WANO GL 2004-01-R1	V.C1.	244
SF15_WANO GL 2004-01-R1_V.C2._15	SF15-01-15	WANO GL 2004-01-R1	V.C2.	245
SF15_WANO GL 2004-01-R1_VI.C3._15	SF15-05-15	WANO GL 2004-01-R1	VI.C3.	252
SF15_WANO GL 2004-01-R1_VII.C2._15	SF15-01-15	WANO GL 2004-01-R1	VII.C2.	253
SF15_WANO GL 2004-01-R1_VII.C2._15	SF15-01-15	WANO GL 2004-01-R1	VII.C2.	254
SF09_SF9 RT 2015_5.3.1_15	SF09-01-15	SF9 RT 2015	5.3.1	255

Table 50: Micro-gaps with Safety Improvements Considered Unnecessary to Implement as Part of IIP

Gap #	SF01_ASME B31.1_MISC_15
Document ID	ASME B31.1
Article/Clause	MISC
Requirement Assessed	Summary of SBR findings - changes to ASME B31.1 from 2007 to 2011 The Safety Basis Report (SBR) presents findings of a review against changes made to ASME B31.1 in the period from 2004 to 2011 [4]. An Updated Code Reconciliation Report "ANSI/ASME Code Reconciliation for Replacement Material, Parts, and Components" [6] was issued to Bruce Power on 2012 December 7, which reviewed and assessed the changes to several codes.
Macro-Gap	SF01-17-15
Issue/Gap Description	Potential issues mentioned in the SBR regarding changes to ASME B31.1 from 2007 to 2011 have not been addressed.
Rationale	Bruce Power complies with the CSA N285.0-12, per existing PROL, which references the applicable ASME Codes. Hence, any code changes and associated impacts are already being addressed as part of the current engineering governance and PROL compliance. Bruce Power is addressing legacy ASME Code changes on the current plant design through the Legacy Registration Project. Follow-up to the Legacy Registration Project documentation and field implementation initiatives have also been included in the IIP. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_ASME BPVC Section III_MISC_15
Document ID	ASME BPVC Section III
Article/Clause	MISC
Requirement Assessed	Changes to ASME Section III from 2011 to 2014 Significant changes to ASME Section III up to the 2013 annual addenda are summarized in "ANSI/ASME Code Reconciliation for Replacement Material, Parts, and Components" [6] Appendix 6, with changes to the material requirements in Appendix 9.
Macro-Gap	SF01-15-15
Issue/Gap Description	There is no evidence that pressure boundary design governance documentation and safety margins has been reviewed for impact of changes in Stress Limits, Bolting Sm Values, Stress Indices for Straight Pipe, Branch Connections and Load Limit values
Rationale	Bruce Power complies with the CSA N285.0-12, per existing PROL, which references the applicable ASME Codes. Hence, any code changes and associated impacts are already being addressed as part of the current engineering governance and PROL compliance. Bruce Power is addressing legacy ASME Code changes on the current plant design through the Legacy Registration Project. Follow-up to the Legacy Registration Project documentation and field implementation initiatives have also been included in the IIP. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_ASME BPVC Section III_MISC_16
Document ID	ASME BPVC Section III
Article/Clause	MISC
Requirement Assessed	Changes to ASME Section III from 2011 to 2014 Significant changes to ASME Section III up to the 2013 annual addenda are summarized in "ANSI/ASME Code Reconciliation for Replacement Material, Parts, and Components" [6] Appendix 6, with changes to the material requirements in Appendix 9.
Macro-Gap	SF01-17-16
Issue/Gap Description	No evidence that pressure boundary design governance documentation and safety margins has been reviewed.
Rationale	Bruce Power complies with the CSA N285.0-12, per existing PROL, which references the applicable ASME Codes. Hence, any code changes and associated impacts are already being addressed as part of the current engineering governance and PROL compliance. Bruce Power is addressing legacy ASME Code changes on the current plant design through the Legacy Registration Project. Follow-up to the Legacy Registration Project documentation and field implementation initiatives have also been included in the IIP. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_ASME BPVC Section VIII_MISC_15
Document ID	ASME BPVC Section VIII
Article/Clause	MISC
Requirement Assessed	Changes to ASME Section VIII from 2011 to 2014
Macro-Gap	SF01-16-15
Issue/Gap Description	The impact on pressure boundary design governance due to changes to bellow design requirements has not been assessed.
Rationale	Bruce Power complies with the CSA N285.0-12, per existing PROL, which references the applicable ASME Codes. Hence, any code changes and associated impacts are already being addressed as part of the current engineering governance and PROL compliance. Bruce Power is addressing legacy ASME Code changes on the current plant design through the Legacy Registration Project. Follow-up to the Legacy Registration Project documentation and field implementation initiatives have also been included in the IIP. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_ASME BPVC Section VIII_MISC_16
Document ID	ASME BPVC Section VIII
Article/Clause	MISC
Requirement Assessed	Changes to ASME Section VIII from 2011 to 2014
Macro-Gap	SF01-18-16
Issue/Gap Description	Pressure boundary design governance documentation and safety margins have not been reviewed for impact of new requirements
Rationale	Bruce Power complies with the CSA N285.0-12, per existing PROL, which references the applicable ASME Codes. Hence, any code changes and associated impacts are already being addressed as part of the current engineering governance and PROL compliance. Bruce Power is addressing legacy ASME Code changes on the current plant design through the Legacy Registration Project. Follow-up to the Legacy Registration Project documentation and field implementation initiatives have also been included in the IIP. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_7.11_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.11 Guaranteed shutdown state
Requirement Assessed	The design authority shall define the guaranteed shutdown state (GSS) that will support safe maintenance activities of the NPP. The design shall provide two independent means of preventing recriticality from any pathway or mechanism when the reactor is in the GSS. The shutdown margin for GSS shall be such that the core will remain subcritical for any credible changes in the core configuration and reactivity addition. Where possible, this shall be achieved without operator intervention. Guidance A GSS is where the reactor remains in a stable, sub-critical state, independent of any perturbation in reactivity produced by any change in core configuration, core properties, or process system failure. The design should describe the GSSs that are expected to be used over the life of the facility, including steps for GSS placement and removal, and functional tests to be performed.
Macro-Gap	SF01-08-15
Issue/Gap Description	Typically the test frequency is determined by safety assessments, probabilistic risk assessment and unavailability analyses. The functional tests to be performed are not reflected in the design documentation; therefore this is assessed as a gap (Gap).
Rationale	This is not considered to be a necessary improvement because the test frequencies are already documented formally in operating documents (e.g., SSTs) as opposed to design documentation. It should be noted that test scope, extent and frequencies are updated based on system and equipment performance and OPEX which requires updates of operating documents. Repeating the same requirements in design documentation may result in unnecessary duplication and potentially an error-likely situation due to having different information on the same requirement in different sources as a result of the timing of their updates. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_7.11_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.11 Guaranteed shutdown state
Requirement Assessed	The design authority shall define the guaranteed shutdown state (GSS) that will support safe maintenance activities of the NPP. The design shall provide two independent means of preventing recriticality from any pathway or mechanism when the reactor is in the GSS. The shutdown margin for GSS shall be such that the core will remain subcritical for any credible changes in the core configuration and reactivity addition. Where possible, this shall be achieved without operator intervention. Guidance A GSS is where the reactor remains in a stable, sub-critical state, independent of any perturbation in reactivity produced by any change in core configuration, core properties, or process system failure. The design should describe the GSSs that are expected to be used over the life of the facility, including steps for GSS placement and removal, and functional tests to be performed.
Macro-Gap	SF01-08-16
Issue/Gap Description	Typically the test frequency is determined by safety assessments, probabilistic risk assessment and unavailability analyses. The functional tests to be performed on the equipment associated with GSS (e.g., auxiliary pumps needed to run for poison, sampling and recirculation) are not reflected in the design documentation; therefore this is assessed as a gap (Gap).
Rationale	This is not considered to be a necessary improvement because the test frequencies are already documented formally in operating documents (e.g., SSTs) as opposed to design documentation. It should be noted that test scope, extent and frequencies are updated based on system and equipment performance and OPEX which requires updates of operating documents. Repeating the same requirements in design documentation may result in unnecessary duplication and potentially an error-likely situation due to having different information on the same requirement in different sources as a result of the timing of their updates. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_7.13_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.13 Seismic qualification and design
Requirement Assessed	The seismic qualification of all SSCs shall meet the requirements of Canadian national or equivalent standards. The design shall include instrumentation for monitoring seismic activity at the site for the life of the plant.
Macro-Gap	SF01-15-16
Issue/Gap Description	Earthquake monitoring instrumentation should be installed in the plant to provide accurate earthquake records to confirm that the plant is fit for continued operation following an earthquake (Gap).
Rationale	The off-site monitoring is an acceptable method per Clause 6.5.2 (c) of N289.1-08 which places the responsibility on the nuclear operator that the means are in place to declare a seismic event through “on-site instrument records, off site seismic data or rapid notifications from appropriate agencies such as the Geological Survey of Canada”. Procedure DPT-PDE-00017, Bruce Power Seismic Qualification Standard includes CSA N289.5 as a basis for seismic qualification (clause 4.1, second paragraph), but notes in Section 4.6 (Post Seismic Response) that notification of an earthquake of magnitude 5 or greater within 500 km of the site will be received from the Southern Ontario Seismograph Network, which has one monitoring station within 20 km of the Bruce site. This is also included in the operating procedures and has been accepted by the CNSC through the acceptance of the procedure noted above, which documents this monitoring approach. The notifications from the Geological Survey of Canada (GSC) are based on data from the National Teleseismic Network supplemented by the Southern Ontario Seismic Network. The GSC data are expert reviewed at GSC Ottawa before being reported. The GSC reporting is first an oral report within 24h of the seismic event if the seismic event exceeds the maximum acceleration for the Design Basis earthquake, otherwise within a business day, followed by written event notification. In summary, Bruce Power complies with the intent of Clause 6.5 (and thus with REGDOC-2.5.2 Clause 7.13) by means of off-site monitoring, which is an acceptable alternative accepted by the CNSC. In this context, gaps associated with seismic instrumentation are specific to address the data collected using the monitoring instruments installed at different floor levels and hence not applicable based on the current methodology used by Bruce Power. Since the post-seismic event notification to the operating staff is considered to be adequate and has been accepted by the CNSC, it is judged that the free field motion accelerometer and placing accelerometers on structures and equipment is not essential. However, installation of in-plant monitoring equipment and developing the

	in-house expertise to interpret and assess earthquake data will be reviewed to establish the feasibility of implementing an in-plant monitoring network and make recommendations in alignment with the progress of the major station refurbishments made under the MCR project. Although the reporting requirements are listed in BP-PROC-00059 Event Response and Reporting, the post-seismic reporting and recording requirements will also be included in DPT-PDE-00017 and NK29-AIM-03600.1-25 as part of periodic updates of these procedures. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_7.6.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.3 Fail-safe design
Requirement Assessed	The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety. To the greatest extent practicable, the application of this principle shall enable plant systems to pass into a safe state if a system or component fails, with no necessity for any action to be taken. Guidance Knowing the failure modes of SSCs is important in applying the fail-safe concept to SSCs important to safety. An analysis, such as a failure modes and effects analysis, should be performed so as to identify the potential failure modes of SSCs important to safety. Failures of SSCs important to safety should be detectable by periodic testing, or revealed by alarms or another reliable indication.
Macro-Gap	SF01-01-15
Issue/Gap Description	As presented in Section 6.1.3 of Part 2 of the Safety Report, to provide a high degree of assurance that a special safety system will perform as designed when called upon to do so, the unavailability target of each is limited to less than 1E-3 yr/yr. In addition, where such choice is available, special safety system components are designed such that the most likely failure modes are in the failsafe direction. It is recognized that in the original design this approach has been followed to the extent practicable. Since there are exceptions to this design rule (e.g., as documented in Design Guide Supplements NK29-DGS-29-03650-003, NK29-DGS-29-03650-004, NK29-DGS-29-03650-004-007 etc.) this is assessed as a gap (Gap).
Rationale	The current design adapts this principle to the extent practicable. Component failures are already addressed in the DSA and PSA and compliance with the dose acceptance criteria is demonstrated. Therefore, operation with such ’exceptions to fail-safe operation’ does not impact plant safety in any significant manner and hence do not warrant design changes. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_7.6.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.3 Fail-safe design
Requirement Assessed	The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety. To the greatest extent practicable, the application of this principle shall enable plant systems to pass into a safe state if a system or component fails, with no necessity for any action to be taken. Guidance Knowing the failure modes of SSCs is important in applying the fail-safe concept to SSCs important to safety. An analysis, such as a failure modes and effects analysis, should be performed so as to identify the potential failure modes of SSCs important to safety. Failures of SSCs important to safety should be detectable by periodic testing, or revealed by alarms or another reliable indication.
Macro-Gap	SF01-20-16
Issue/Gap Description	As presented in Section 6.1.3 of Part 2 of the Safety Report, to provide a high degree of assurance that a special safety system will perform as designed when called upon to do so, the unavailability target of each is limited to less than 1E-3 yr/yr. In addition, where such choice is available, special safety system components are designed such that the most likely failure modes are in the failsafe direction. It is recognized that in the original design this approach has been followed to the extent practicable. Since there are exceptions to this design rule (e.g., as documented in Design Guide Supplements NK29-DGS-29-03650-003, NK29-DGS-29-03650-004, NK29-DGS-29-03650-004-007 etc.) this is assessed as a gap (Gap).
Rationale	The current design adapts this principle to the extent practicable. Component failures are already addressed in the DSA and PSA and compliance with the dose acceptance criteria is demonstrated. Therefore, operation with such 'exceptions to fail-safe operation' does not impact plant safety in any significant manner and hence do not warrant design changes. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_8.13.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13.3 Radiation monitoring
Requirement Assessed	Equipment shall be provided to ensure that there is adequate radiation monitoring in operational states, DBAs and DECs. Stationary alarming dose rate meters shall be provided: 1. for monitoring the local radiation dose rate at places routinely occupied by operating personnel 2. where the changes in radiation levels may be such that access may be limited for periods of time 3. to indicate, automatically and in real-time, the general radiation level at appropriate locations in operational states, DBAs and DECs 4. to give sufficient information in the control room or at the appropriate control location for operational states, DBAs and DECs, to enable plant personnel to initiate corrective actions when necessary Monitors shall be provided for measuring the activity of radioactive substances in the atmosphere: 1. for areas routinely occupied by personnel 2. for areas where the levels of activity of airborne radioactive materials may, on occasion, be expected to necessitate protective measures 3. to give an indication in the control room, or in other appropriate locations, of when a high concentration of radionuclides is detected Facilities shall be provided for monitoring individual doses to and contamination of personnel. Stationary equipment and laboratory facilities shall be provided to determine the concentration of selected radionuclides in fluid process systems as appropriate, and in gas and liquid samples taken from plant systems or the environment. Stationary equipment shall be provided for monitoring the effluents prior to or during discharge to the environment.
Macro-Gap	SF01-14-15
Issue/Gap Description	Compliance with the requirement for radiation monitoring equipment that indicate automatically and in real time the radiation levels cannot be confirmed in the design documentation. Therefore, this is assessed as a gap (Gap).

Rationale	Bruce Power radiation monitoring equipment and the associated operating documentation such as AIMs, EOPs are adequate to respond to accidents involving radioactivity releases. This gap is not considered to be safety significant. This gap, based on Bruce Power's input, is categorized as "Closed" in the PSR database and any follow-up action(s) for its implementation or oversight is documented including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_8.13.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13.3 Radiation monitoring
Requirement Assessed	Equipment shall be provided to ensure that there is adequate radiation monitoring in operational states, DBAs and DECs. Stationary alarming dose rate meters shall be provided: 1. for monitoring the local radiation dose rate at places routinely occupied by operating personnel 2. where the changes in radiation levels may be such that access may be limited for periods of time 3. to indicate, automatically and in real-time, the general radiation level at appropriate locations in operational states, DBAs and DECs 4. to give sufficient information in the control room or at the appropriate control location for operational states, DBAs and DECs, to enable plant personnel to initiate corrective actions when necessary Monitors shall be provided for measuring the activity of radioactive substances in the atmosphere: 1. for areas routinely occupied by personnel 2. for areas where the levels of activity of airborne radioactive materials may, on occasion, be expected to necessitate protective measures 3. to give an indication in the control room, or in other appropriate locations, of when a high concentration of radionuclides is detected Facilities shall be provided for monitoring individual doses to and contamination of personnel. Stationary equipment and laboratory facilities shall be provided to determine the concentration of selected radionuclides in fluid process systems as appropriate, and in gas and liquid samples taken from plant systems or the environment. Stationary equipment shall be provided for monitoring the effluents prior to or during discharge to the environment.
Macro-Gap	SF01-14-16
Issue/Gap Description	Compliance with the requirement for radiation monitoring equipment that indicate automatically and in real time the radiation levels cannot be confirmed in the design documentation. Therefore, this is assessed as a gap (Gap).

Rationale	Bruce Power radiation monitoring equipment and the associated operating documentation such as AIMs, EOPs are adequate to respond to accidents involving radioactivity releases. This gap is not considered to be safety significant. This gap, based on Bruce Power's input, is categorized as "Closed" in the PSR database and any follow-up action(s) for its implementation or oversight is documented including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_8.9.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.9.2 DC and uninterruptible power systems
Requirement Assessed	The design of the direct current (DC) power systems and uninterruptible AC power systems (if applicable) shall specify operating mission times when performing the intended safety functions of the connected loads and meet the capacity requirements of section 7.10. The design shall include provisions for periodic testing for DC power and uninterruptible AC power supplies to confirm their capability. Guidance DC power systems DC power systems important to safety should be designed to be independent of the effects of DBAs to which they must respond, and be fully functional during and following such accidents. Redundant load groups should each have a DC power supply division consisting of one or more batteries, one or more battery chargers, distribution system, protection and isolation features. Each DC power supply division should be independent and physically separate from other DC divisions. Battery chargers should be designed to prevent transients on the AC supply from affecting the functioning of the DC system, and from DC transients affecting the AC supply. Uninterruptible AC power systems Uninterruptible AC power systems important to safety should be designed to be independent of the effects of design-basis accidents to which they must respond, and be fully functional during and following such accidents. Each division of uninterruptible AC power system should consist of: • an AC power supply and a DC power supply to an inverter • a separate AC power supply from the same division • a feature to automatically switch between the inverter output and the separate AC supply The electrical characteristics and requirements of the connected loads should be considered in the design so that interactions with the uninterruptible AC power system do not degrade the safety support functions of the loads supplied.

	Uninterruptible AC power systems should be designed to prevent transients on the AC supply to the battery charger or on the DC supply to the inverter from affecting the functioning of the inverter.
Macro-Gap	SF01-12-15
Issue/Gap Description	Operational Safety Requirements for Bruce A Electrical Systems [NK21-OSR-53000/55000-0001, R000] present the safety limits, applicable analysis and surveillance requirements for Bruce A Electrical Power Systems. Since the capacity requirements and the design provisions for periodic testing as required in Clause 7.10 are not sufficiently documented, this is assessed as a gap. (Gap)
Rationale	Periodic testing for DC power and uninterruptible AC power supplies to confirm their capability is governed by the Equipment Reliability Program and as such necessary provisions are in place. Such requirements need not be included in the design documentation because the test scope, extent and frequencies are updated based on system and equipment performance and OPEX. Repeating the same requirements in design documentation may result in unnecessary duplication and potentially an error-likely situation due to having different information on the same requirement in different sources as a result of the timing of their updates. This gap, based on Bruce Power's input, is categorized as "Closed" in the PSR database and any follow-up action(s) for its implementation or oversight is documented including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_8.9.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.9.2 DC and uninterruptible power systems
Requirement Assessed	The design of the direct current (DC) power systems and uninterruptible AC power systems (if applicable) shall specify operating mission times when performing the intended safety functions of the connected loads and meet the capacity requirements of section 7.10. The design shall include provisions for periodic testing for DC power and uninterruptible AC power supplies to confirm their capability. Guidance DC power systems DC power systems important to safety should be designed to be independent of the effects of DBAs to which they must respond, and be fully functional during and following such accidents. Redundant load groups should each have a DC power supply division consisting of one or more batteries, one or more battery chargers, distribution system, protection and isolation features. Each DC power supply division should be independent and physically separate from other DC divisions. Battery chargers should be designed to prevent transients on the AC supply from affecting the functioning of the DC system, and from DC transients affecting the AC supply. Uninterruptible AC power systems Uninterruptible AC power systems important to safety should be designed to be independent of the effects of design-basis accidents to which they must respond, and be fully functional during and following such accidents. Each division of uninterruptible AC power system should consist of: • an AC power supply and a DC power supply to an inverter • a separate AC power supply from the same division • a feature to automatically switch between the inverter output and the separate AC supply The electrical characteristics and requirements of the connected loads should be considered in the design so that interactions with the uninterruptible AC power system do not degrade the safety support functions of the loads supplied.

	Uninterruptible AC power systems should be designed to prevent transients on the AC supply to the battery charger or on the DC supply to the inverter from affecting the functioning of the inverter.
Macro-Gap	SF01-12-16
Issue/Gap Description	Operational Safety Requirements for Bruce B Electrical Systems [NK29-OSR-53000-55000-00001, R000] present the safety limits, applicable analysis and surveillance requirements for Bruce B Electrical Power Systems. Since the capacity requirements and the design provisions for periodic testing as required in Clause 7.10 are not sufficiently documented, this is assessed as a gap. (Gap)
Rationale	Periodic testing for DC power and uninterruptible AC power supplies to confirm their capability is governed by the Equipment Reliability Program and as such necessary provisions are in place. Such requirements need not be included in the design documentation because the test scope, extent and frequencies are updated based on system and equipment performance and OPEX. Repeating the same requirements in design documentation may result in unnecessary duplication and potentially an error-likely situation due to having different information on the same requirement in different sources as a result of the timing of their updates. This gap, based on Bruce Power's input, is categorized as "Closed" in the PSR database and any follow-up action(s) for its implementation or oversight is documented including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_8.9_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.9 Electrical power systems
Requirement Assessed	The design shall specify the required functions and performance characteristics of each electrical power system that provides normal, standby, emergency and alternate power supplies to ensure: 1. sufficient capacity to support the safety functions of the connected loads in operational states, DBAs and DECs 2. availability and reliability is commensurate with the safety significance of the connected loads The requirements of both the standby and emergency power systems may be met by a single system. Electrical power systems shall be designed to include the various modes of interaction between offsite power and onsite power. In addition, design provisions shall be established for coping with grid disturbances including conditions caused by solar flare (coronal mass ejection) events. The design shall specify: 1. environmental and electromagnetic conditions to which electrical equipment and cables may be subjected 2. limits on electromagnetic emissions conducted or radiated from electrical equipment The electrical power systems shall include appropriate protection, control, monitoring and testing facilities. Guidance A systematic approach should be followed to identify the electrical power systems needed in order to ensure that SSCs necessary to fulfill the safety functions are powered from electrical power supplies with appropriate safety classification and reliability. The design bases, design criteria, regulatory documents, standards, and other documents that will be used to design the electrical power systems should be specified. For each of the electrical power systems, the design bases include: • consideration of all modes of operation, plant states up to DECs and all credible events that could impact the electrical power systems

	• reliability and availability targets for systems and key equipment • capacity and performance requirements • identification of all loads (i.e., the systems and equipment that require electric power to perform their safety functions) including electrical characteristics, maximum demand conditions, and safety classification • protective schemes and coordination of protection • specification of acceptable ranges of voltage and frequency for continuous operation of the connected loads for each electrical power system • identification of acceptable ranges for onsite and offsite transient disturbance events that could impact electrical power systems The design should specify the requirements for the preferred power supply (PPS) (i.e., the normal alternating current (AC) power supplies for plant electrical systems important to safety) and the plant interface with the transmission grid to reduce the potential for loss of normal AC power supplies. Transmission system studies should be undertaken for reasonably expected grid system conditions and disturbances to demonstrate that normal AC power supplies will not be degraded to a level that causes unnecessary challenges to safety systems, standby and emergency power supply systems. Performance criteria should be established for: • unit generator performance during defined frequency and voltage excursions to ensure that generators remain connected to the electrical grid • lightning and surge protection design provisions to protect the plant electrical distribution systems against transient over-voltage conditions such as switching and lightning surges The normal AC electrical power systems should have the capacity and capability to supply all plant electrical loads during operational states, DBAs and DECs. Normal AC power supplies should be designed to: • prevent deviations from normal operation • prevent single failures from impacting more than one redundant division of electrical power supply • avoid preventable challenges to standby and emergency systems as a result of an electrical system disturbance, transient, or upset condition (e.g., turbine-generator trip) Electrical power supply from the offsite power system to the onsite power system should be supplied by a minimum of two physically independent transmission lines designed and located in order to minimize the likelihood of their simultaneous failure. The safety analysis should provide

	information concerning offsite power circuits coming from the transmission system to the plant switchyard. A switchyard common to both circuits is acceptable, but separate transmission line towers should be used. For some reactor designs, it might be sufficient to have only one offsite power connection, although this should be justified. Each of the plant's offsite transmission lines should have the capacity and capability to supply power to all plant electrical loads under all plant states. A minimum of one offsite transmission line and associated PPS should be designed to be automatically available to provide power to its associated safety divisions within a few seconds following an AOO or a DBA. A second PPS circuit should be designed to be available within a period of time commensurate with the requirement to support plant safety functions during AOOs and DBAs. For plants designed for house load operation, the normal AC power system should be designed to accommodate generator voltage and frequency transients associated with transferring from normal operation to the house load operating mode.
Macro-Gap	SF01-12-15
Issue/Gap Description	There is no design limit specified on electromagnetic emissions conducted or radiated from electrical equipment. Therefore, this is assessed as a gap (Gap).
Rationale	Historical specifications and testing of I&C equipment by Ontario Hydro and AECL made use of a combination of internal and military level specifications. EMC specifications and testing covered for areas such as Electro-Static Discharge, Radiated Interference, Transient Interference Immunity and others were prescribed. In addition, military level tests for equipment such as neutronic amplifiers were utilized such as MIL-STD-461A, 461 (Methods and Levels). Bruce Power currently makes use of IEC 61000 series standards for susceptibility and emissions level testing of I&C equipment. B-DG-50000-00001, "Electrical and I&C Design Guides, Standards and Aids" provides guidance on use of Electro-Magnetic Compatibility (EMC) procurement standards as identified in B-REP-60000-00002 "EMC Procurement Specification Report For I&C Equipment". Working in conjunction with EPRI in 2012 Bruce Power adapted an industry graded approach for EMC testing, both susceptibility and emission, for different classifications of Safety-Related, Important to Safety and Non-Safety Related levels. The prescribed tests and levels were derived from the industry standard TR-102323, Rev 3 of which makes use of commercial IEC 61000 series and military tests. Hence, this gap is not considered to be safety significant.

	This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CNSC REGDOC 2.5.2_8.9_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.9 Electrical power systems
Requirement Assessed	The design shall specify the required functions and performance characteristics of each electrical power system that provides normal, standby, emergency and alternate power supplies to ensure: 1. sufficient capacity to support the safety functions of the connected loads in operational states, DBAs and DECs 2. availability and reliability is commensurate with the safety significance of the connected loads The requirements of both the standby and emergency power systems may be met by a single system. Electrical power systems shall be designed to include the various modes of interaction between offsite power and onsite power. In addition, design provisions shall be established for coping with grid disturbances including conditions caused by solar flare (coronal mass ejection) events. The design shall specify: 1. environmental and electromagnetic conditions to which electrical equipment and cables may be subjected 2. limits on electromagnetic emissions conducted or radiated from electrical equipment The electrical power systems shall include appropriate protection, control, monitoring and testing facilities. Guidance A systematic approach should be followed to identify the electrical power systems needed in order to ensure that SSCs necessary to fulfill the safety functions are powered from electrical power supplies with appropriate safety classification and reliability. The design bases, design criteria, regulatory documents, standards, and other documents that will be used to design the electrical power systems should be specified. For each of the electrical power systems, the design bases include: • consideration of all modes of operation, plant states up to DECs and all credible events that could impact the electrical power systems

	• reliability and availability targets for systems and key equipment • capacity and performance requirements • identification of all loads (i.e., the systems and equipment that require electric power to perform their safety functions) including electrical characteristics, maximum demand conditions, and safety classification • protective schemes and coordination of protection • specification of acceptable ranges of voltage and frequency for continuous operation of the connected loads for each electrical power system • identification of acceptable ranges for onsite and offsite transient disturbance events that could impact electrical power systems The design should specify the requirements for the preferred power supply (PPS) (i.e., the normal alternating current (AC) power supplies for plant electrical systems important to safety) and the plant interface with the transmission grid to reduce the potential for loss of normal AC power supplies. Transmission system studies should be undertaken for reasonably expected grid system conditions and disturbances to demonstrate that normal AC power supplies will not be degraded to a level that causes unnecessary challenges to safety systems, standby and emergency power supply systems. Performance criteria should be established for: • unit generator performance during defined frequency and voltage excursions to ensure that generators remain connected to the electrical grid • lightning and surge protection design provisions to protect the plant electrical distribution systems against transient over-voltage conditions such as switching and lightning surges The normal AC electrical power systems should have the capacity and capability to supply all plant electrical loads during operational states, DBAs and DECs. Normal AC power supplies should be designed to: • prevent deviations from normal operation • prevent single failures from impacting more than one redundant division of electrical power supply • avoid preventable challenges to standby and emergency systems as a result of an electrical system disturbance, transient, or upset condition (e.g., turbine-generator trip) Electrical power supply from the offsite power system to the onsite power system should be supplied by a minimum of two physically independent transmission lines designed and located in order to minimize the likelihood of their simultaneous failure. The safety analysis should provide

	information concerning offsite power circuits coming from the transmission system to the plant switchyard. A switchyard common to both circuits is acceptable, but separate transmission line towers should be used. For some reactor designs, it might be sufficient to have only one offsite power connection, although this should be justified. Each of the plant's offsite transmission lines should have the capacity and capability to supply power to all plant electrical loads under all plant states. A minimum of one offsite transmission line and associated PPS should be designed to be automatically available to provide power to its associated safety divisions within a few seconds following an AOO or a DBA. A second PPS circuit should be designed to be available within a period of time commensurate with the requirement to support plant safety functions during AOOs and DBAs. For plants designed for house load operation, the normal AC power system should be designed to accommodate generator voltage and frequency transients associated with transferring from normal operation to the house load operating mode.
Macro-Gap	SF01-12-16
Issue/Gap Description	There is no design limit specified on electromagnetic emissions conducted or radiated from electrical equipment. Therefore, this is assessed as a gap (Gap).
Rationale	Historical specifications and testing of I&C equipment by Ontario Hydro and AECL made use of a combination of internal and military level specifications. EMC specifications and testing covered for areas such as Electro-Static Discharge, Radiated Interference, Transient Interference Immunity and others were prescribed. In addition, military level tests for equipment such as neutronic amplifiers were utilized such as MIL-STD-461A, 461 (Methods and Levels). Bruce Power currently makes use of IEC 61000 series standards for susceptibility and emissions level testing of I&C equipment. B-DG-50000-00001, "Electrical and I&C Design Guides, Standards and Aids" provides guidance on use of Electro-Magnetic Compatibility (EMC) procurement standards as identified in B-REP-60000-00002 "EMC Procurement Specification Report For I&C Equipment". Working in conjunction with EPRI in 2012 Bruce Power adapted an industry graded approach for EMC testing, both susceptibility and emission, for different classifications of Safety-Related, Important to Safety and Non-Safety Related levels. The prescribed tests and levels were derived from the industry standard TR-102323, Rev 3 of which makes use of commercial IEC 61000 series and military tests. Hence, this gap is not considered to be safety significant.

	This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CSA N289.1_6.5.6.3_16
Document ID	CSA N289.1
Article/Clause	6.5.6.3
Requirement Assessed	Clause 6.5.6.3 states: "Operator response shall be based on accurate earthquake records. All significant earthquake data, including intensity and duration, shall be recorded to (a) account for loss of service life (fatigue usage factor) due to seismically induced stress cycles; and (b) aid in determination of the need to shut down or continue operation of the plant".
Macro-Gap	SF01-15-16
Issue/Gap Description	Current governing documents do not address the need for recording equipment to be installed in the plant to satisfy the intent of clause 6.5.6.
Rationale	The off-site monitoring is an acceptable method per Clause 6.5.2 (c) of N289.1-08 which places the responsibility on the nuclear operator that the means are in place to declare a seismic event through "on-site instrument records, off site seismic data or rapid notifications from appropriate agencies such as the Geological Survey of Canada". Procedure DPT-PDE-00017, Bruce Power Seismic Qualification Standard includes CSA N289.5 as a basis for seismic qualification (clause 4.1, second paragraph), but notes in Section 4.6 (Post Seismic Response) that notification of an earthquake of magnitude 5 or greater within 500 km of the site will be received from the Southern Ontario Seismograph Network, which has one monitoring station within 20 km of the Bruce site. This is also included in the operating procedures and has been accepted by the CNSC through the acceptance of the procedure noted above, which documents this monitoring approach. The notifications from the Geological Survey of Canada (GSC) are based on data from the National Teleseismic Network supplemented by the Southern Ontario Seismic Network. The GSC data are expert reviewed at GSC Ottawa before being reported. The GSC reporting is first an oral report within 24h of the seismic event if the seismic event exceeds the maximum acceleration for the Design Basis earthquake, otherwise within a business day, followed by written event notification. In summary, Bruce Power complies with the intent of Clause 6.5 by means of off-site monitoring, which is an acceptable alternative accepted by the CNSC. In this context, gaps associated with seismic instrumentation are specific to address the data collected using the monitoring instruments installed at different floor levels and hence not applicable based on the current methodology used by Bruce Power. Since the post-seismic event notification to the operating staff is considered to be adequate and has been accepted by the CNSC, it is judged that the free field motion accelerometer and placing accelerometers on structures and equipment is not essential. However, installation of in-plant monitoring equipment and developing the

	in-house expertise to interpret and assess earthquake data will be reviewed to establish the feasibility of implementing an in-plant monitoring network and make recommendations in alignment with the progress of the major station refurbishments made under the MCR project. Although the reporting requirements are listed in BP-PROC-00059 Event Response and Reporting, the post-seismic reporting and recording requirements will also be included in DPT-PDE-00017 and NK29-AIM-03600.1-25 as part of periodic updates of these procedures. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CSA N289.1_6.5.6.4_16
Document ID	CSA N289.1
Article/Clause	6.5.6.4
Requirement Assessed	Clause 6.5.6.3 states: "The data collected during the earthquake from earthquake monitoring instruments installed at different levels in the plant shall be compared to the design floor response spectra to assess if the design stress level was exceeded. The plant shall be shut down for inspection when the measured floor response spectra exceed the design floor response spectra. Local high acceleration spikes at high frequencies in the measured ground or floor response spectra shall be evaluated, but generally do not constitute design basis exceedance, as they usually have low-energy content. Where it is determined that the earthquake intensity has reached the design basis level, the plant shall be shut down for inspection."
Macro-Gap	SF01-15-16
Issue/Gap Description	Clause 6.5.6.4 requires data collected from monitoring instruments installed at different levels in the plant to be compared with the design floor response spectra to assess if the design stress levels have been exceeded. This type of monitoring instrumentation would quickly confirm that the plant is fit for continued operation following an earthquake.
Rationale	The off-site monitoring is an acceptable method per Clause 6.5.2 (c) of N289.1-08 which places the responsibility on the nuclear operator that the means are in place to declare a seismic event through "on-site instrument records, off site seismic data or rapid notifications from appropriate agencies such as the Geological Survey of Canada". Procedure DPT-PDE-00017, Bruce Power Seismic Qualification Standard includes CSA N289.5 as a basis for seismic qualification (clause 4.1, second paragraph), but notes in Section 4.6 (Post Seismic Response) that notification of an earthquake of magnitude 5 or greater within 500 km of the site will be received from the Southern Ontario Seismograph Network, which has one monitoring station within 20 km of the Bruce site. This is also included in the operating procedures and has been accepted by the CNSC through the acceptance of the procedure noted above, which documents this monitoring approach. The notifications from the Geological Survey of Canada (GSC) are based on data from the National Teleseismic Network supplemented by the Southern Ontario Seismic Network. The GSC data are expert reviewed at GSC Ottawa before being reported. The GSC reporting is first an oral report within 24h of the seismic event if the seismic event exceeds the maximum acceleration for the Design Basis earthquake, otherwise within a business day, followed by written event notification. In summary, Bruce Power complies with the intent of Clause 6.5 by means of off-site monitoring, which is an acceptable alternative accepted by the CNSC. In this context, gaps associated with seismic instrumentation are specific to address the data collected using the monitoring instruments

	installed at different floor levels and hence not applicable based on the current methodology used by Bruce Power. Since the post-seismic event notification to the operating staff is considered to be adequate and has been accepted by the CNSC, it is judged that the free field motion accelerometer and placing accelerometers on structures and equipment is not essential. However, installation of in-plant monitoring equipment and developing the in-house expertise to interpret and assess earthquake data will be reviewed to establish the feasibility of implementing an in-plant monitoring network and make recommendations in alignment with the progress of the major station refurbishments made under the MCR project. Although the reporting requirements are listed in BP-PROC-00059 Event Response and Reporting, the post-seismic reporting and recording requirements will also be included in DPT-PDE-00017 and NK29-AIM-03600.1-25 as part of periodic updates of these procedures. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CSA N290.0-11_4.9-4.13_16
Document ID	CSA N290.0-11
Article/Clause	4.9-4.13
Requirement Assessed	Address the requirements related to safety support systems, pressure-retaining SSCs, instrumentation, control and monitoring, equipment qualification and dynamic piping effects.
Macro-Gap	SF01-05-16
Issue/Gap Description	Gap against clause 4.11.2.13: Bruce A design documentation does not explicitly reflect the requirement to minimize unavailability due to calibration and the time during which an instrument loop is unavailable due to calibration to be included in the unavailability of the loop.
Rationale	The Bruce A SFR1 assessment stated that, although the Bruce A design documentation does not explicitly reflect this requirement, the error models to capture the different sources of errors associated with the calibration of instrumentation are discussed in Part 3 of the Safety Report, and consequently it was assessed as an Acceptable Deviation. Part 3 of the Safety Report is very similar for Bruce A and B in this regard, and thus the assessment for Bruce A applies equally to Bruce B. This gap is considered to be unnecessary to implement, and is categorized as "Closed" in the PSR Database.

Gap #	SF01_CSA N290.2-11_5.12.5_15
Document ID	CSA N290.2-11
Article/Clause	5.12.5
Requirement Assessed	Venting and Draining.
Macro-Gap	SF01-14-15
Issue/Gap Description	Provision of a drain between isolation and check valves where hazardous fluids could be trapped is not explicitly reflected in the design documentation.
Rationale	Bruce Power employs operational means to mitigate potential risk from hazardous fluids that could be trapped between isolation and check valves. In addition Bruce Power will conduct the following review: (1) indicate the lines where hazardous fluids could be trapped during ECI operation, and (2) specify whether there is a drain; if there is no drain, then justification is required as to why it is acceptable. The corresponding sections of the DM should be revised based on the results of such assessment. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_CSA N290.2-11_5.12.5_16
Document ID	CSA N290.2-11
Article/Clause	5.12.5
Requirement Assessed	Venting and Draining.
Macro-Gap	SF01-14-16
Issue/Gap Description	Provision of a drain between isolation and check valves where hazardous fluids could be trapped is not explicitly reflected in the design documentation.
Rationale	Bruce Power employs operational means to mitigate potential risk from hazardous fluids that could be trapped between isolation and check valves. In addition Bruce Power will conduct the following review: (1) indicate the lines where hazardous fluids could be trapped during ECI operation, and (2) specify whether there is a drain; if there is no drain, then justification is required as to why it is acceptable. The corresponding sections of the DM should be revised based on the results of such assessment. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF01_SF1 RT_5.4_16
Document ID	SF1 RT
Article/Clause	5.4
Requirement Assessed	This review task requires a review of the adequacy of the design basis documentation. The purpose of this review task is to ensure that all significant documentation relating to the original design basis has been obtained, securely stored and updated to reflect all the modifications made to the plant and procedures since its commissioning. Refer to section 4.1 for more detailed information.
Macro-Gap	SF01-21-16
Issue/Gap Description	Pressure Boundary Quality Assurance Program: Numerous issues require effective resolution to ensure a robust program and repeat findings from previous audits.
Rationale	An audit was performed in 2014 to verify Bruce Power's compliance with all sections of N285.0-08, excluding Section 18 (Audits) (see section 7.2.4). The audit identified repeat findings indicating that previous activities taken to address the adverse conditions were not successful. Seven areas were considered to be continuing findings, meaning that there were open assignments that had yet to be completed; therefore the adverse conditions still existed. The audit evaluated that BP-PROG-00.04 Revision 20 [129] was not fully compliant in 18 of the 19 sections. Additionally, the audit found that some elements were either not fully implemented, or organizational compliance is such that the defined process may not function as intended. The action to complete the AR associated with the audit findings can be addressed through Bruce Power's current practices. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF03_CSA N289.1_5.3.11_16
Document ID	CSA N289.1
Article/Clause	5.3.11
Requirement Assessed	Clauses 5.2.6 through 5.4 are new clauses added to the 2008 edition that introduce the other standards in the series (i.e., N289.2, N289.3, N289.4) to address requirements for the development of earthquake ground motion, load combinations, seismic qualification, seismic evaluation of existing plants, and design modifications of qualified SSCs. These clauses include requirements for the application of the SMA methodology, for operator response to seismic events, for maintaining seismic qualification, for the seismic evaluation of existing plants and for the design modification of qualified SSCs. The 2014 update includes a new clause (5.3.11) that adds a requirement that states “Each facility shall have a periodic evaluation to demonstrate readiness to cope with the potential consequences of a beyond design basis seismic event”, and a note that states “As a minimum, the evaluation will be carried out once every 10 years” [30].
Macro-Gap	SF03-01-16
Issue/Gap Description	The new requirement added in the 2014 update, clause 5.3.11, for a periodic evaluation every 10 years “to cope with the potential consequences of a beyond design basis seismic event”, is identified as a gap.
Rationale	This is a documentation update gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF03_CSA N289.1_6.5.6.3_16
Document ID	CSA N289.1
Article/Clause	6.5.6.3
Requirement Assessed	The section on post-seismic recovery (clause 6.5.6) requires that "…a detailed engineering report shall be prepared to document assessments of damage to SSCs…The report shall include all data records from in-plant seismic monitoring systems…to determine whether the seismic design basis of the plant has been exceeded.". Clause 6.5.6.3 states: "Operator response shall be based on accurate earthquake records. All significant earthquake data, including intensity and duration, shall be recorded to (a) account for loss of service life (fatigue usage factor) due to seismically induced stress cycles; and (b) aid in determination of the need to shut down or continue operation of the plant".
Macro-Gap	SF03-02-16
Issue/Gap Description	Current governing documents do not address the need for recording equipment to be installed in the plant.
Rationale	The off-site monitoring is an acceptable method per Clause 6.5.2 (c) of N289.1-08 which places the responsibility on the nuclear operator that the means are in place to declare a seismic event through "on-site instrument records, off site seismic data or rapid notifications from appropriate agencies such as the Geological Survey of Canada". Procedure DPT-PDE-00017, Bruce Power Seismic Qualification Standard includes CSA N289.5 as a basis for seismic qualification (clause 4.1, second paragraph), but notes in Section 4.6 (Post Seismic Response) that notification of an earthquake of magnitude 5 or greater within 500 km of the site will be received from the Southern Ontario Seismograph Network, which has one monitoring station within 20 km of the Bruce site. This is also included in the operating procedures and has been accepted by the CNSC through the acceptance of the procedure noted above, which documents this monitoring approach. The notifications from the Geological Survey of Canada (GSC) are based on data from the National Teleseismic Network supplemented by the Southern Ontario Seismic Network. The GSC data are expert reviewed at GSC Ottawa before being reported. The GSC reporting is first an oral report within 24h of the seismic event if the seismic event exceeds the maximum acceleration for the Design Basis earthquake, otherwise within a business day, followed by written event notification. In summary, Bruce Power complies with the intent of Clause 6.5 by means of off-site monitoring, which is an acceptable alternative accepted by the CNSC. In this context, gaps associated with seismic instrumentation are specific to address the data collected using the monitoring instruments installed at different floor levels and hence not applicable based on the current methodology used by Bruce Power. Since the post-seismic event notification to the operating staff is considered to be adequate and has been accepted by the CNSC, it is judged that the free field motion

	accelerometer and placing accelerometers on structures and equipment is not essential. However, installation of in-plant monitoring equipment and developing the in-house expertise to interpret and assess earthquake data will be reviewed to establish the feasibility of implementing an in-plant monitoring network and make recommendations in alignment with the progress of the major station refurbishments made under the MCR project. Although the reporting requirements are listed in BP-PROC-00059 Event Response and Reporting, the post-seismic reporting and recording requirements will also be included in DPT-PDE-00017 and NK29-AIM-03600.1-25 as part of periodic updates of these procedures. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF03_CSA N289.1_6.5.6.4_16
Document ID	CSA N289.1
Article/Clause	6.5.6.4
Requirement Assessed	The data collected during the earthquake from earthquake monitoring instruments installed at different levels in the plant shall be compared to the design floor response spectra to assess if the design stress level was exceeded.
Macro-Gap	SF03-02-16
Issue/Gap Description	Clause 6.5.6.4 requires data collected from monitoring instruments installed at different levels in the plant to be compared with the design floor response spectra to assess if the design stress levels have been exceeded. This type of monitoring instrumentation would quickly confirm that the plant is fit for continued operation following an earthquake.
Rationale	The off-site monitoring is an acceptable method per Clause 6.5.2 (c) of N289.1-08 which places the responsibility on the nuclear operator that the means are in place to declare a seismic event through “on-site instrument records, off site seismic data or rapid notifications from appropriate agencies such as the Geological Survey of Canada”. Procedure DPT-PDE-00017, Bruce Power Seismic Qualification Standard includes CSA N289.5 as a basis for seismic qualification (clause 4.1, second paragraph), but notes in Section 4.6 (Post Seismic Response) that notification of an earthquake of magnitude 5 or greater within 500 km of the site will be received from the Southern Ontario Seismograph Network, which has one monitoring station within 20 km of the Bruce site. This is also included in the operating procedures and has been accepted by the CNSC through the acceptance of the procedure noted above, which documents this monitoring approach. The notifications from the Geological Survey of Canada (GSC) are based on data from the National Teleseismic Network supplemented by the Southern Ontario Seismic Network. The GSC data are expert reviewed at GSC Ottawa before being reported. The GSC reporting is first an oral report within 24h of the seismic event if the seismic event exceeds the maximum acceleration for the Design Basis earthquake, otherwise within a business day, followed by written event notification. In summary, Bruce Power complies with the intent of Clause 6.5 by means of off-site monitoring, which is an acceptable alternative accepted by the CNSC. In this context, gaps associated with seismic instrumentation are specific to address the data collected using the monitoring instruments installed at different floor levels and hence not applicable based on the current methodology used by Bruce Power. Since the post-seismic event notification to the operating staff is considered to be adequate and has been accepted by the CNSC, it is judged that the free field motion accelerometer and placing accelerometers on structures and equipment is not essential.

	However, installation of in-plant monitoring equipment and developing the in-house expertise to interpret and assess earthquake data will be reviewed to establish the feasibility of implementing an in-plant monitoring network and make recommendations in alignment with the progress of the major station refurbishments made under the MCR project. Although the reporting requirements are listed in BP-PROC-00059 Event Response and Reporting, the post-seismic reporting and recording requirements will also be included in DPT-PDE-00017 and NK29-AIM-03600.1-25 as part of periodic updates of these procedures. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF03_CSA N289.2_4.4.2.2_16
Document ID	CSA N289.2
Article/Clause	4.4.2.2
Requirement Assessed	This section requires the earthquake history of the region to be investigated, justification of earthquake parameters when published information is re-evaluated (clause 4.2.1.2), presentation of the earthquake history in the form of maps and tables with the information compiled in a specific way, Local site related investigations are required by clause 4.3.3.2, which states "…detailed investigations shall be conducted to obtain the information specified in clause 4.3.3.1." and clause 4.3.3.1 states "the main purposes of the site geological investigations are (a) to determine the structural, geological, and tectonic setting of the site in relation to the regional information in order to establish the potential for earthquakes in the site vicinity…". In Section 4.4 investigations of seismically induced phenomena are required, and clause 4.4.2.2 applies to the Bruce site, which states: "for sites on the shore of a confined body of water, an investigation shall be made of the potential for seismic seiches and consequent surges along the shore that could affect the safety and operation of the nuclear power plant."
Macro-Gap	SF03-03-16
Issue/Gap Description	The available documentation does not indicate that an investigation of the potential for a seismic seiche and consequent surges along the shore that could affect the safety of the plant were done.
Rationale	As described in section 5.1.1 of the Bruce A and Bruce B Safety Factor 7 Reports, an extensive screening assessment was conducted based on a screening methodology submitted to CNSC staff in (NK21-CORR-00531-09253/NK29-CORR-00531-09881, Submission of Revised Bruce Power Probabilistic Risk Assessment Guide – Screening and Disposition of External Events, Bruce Power Letter, F. Saunders to R. Lojk, March 9, 2012). The list of potential external hazards considered is provided in Table 6. This list covers all the external hazards outlined in Section 5.1 as well as several hazards that could be classified as internal hazards. These hazards were initially subjected to a first-level screening (NK21-CORR-00531-09809/NK29-CORR-00531-10287, Bruce A and B External Hazard Assessment, Bruce Power Letter, F. Saunders to R. Lojk, September 28, 2012), and the hazards which were not eliminated in the first level were then subjected to a second level of screening ([NK21-CORR-00531-10848/NK29-CORR-00531-11226, Bruce A and B External Hazard Assessment, Bruce Power Letter, F. Saunders to R. Lojk, December 12, 2013]). Following this second level of screening, the only hazards requiring assessment are tornadoes, high winds and external flooding. This gap is considered not to be safety significant. Further investigation of

	the available documentation in the current records system can be performed to establish the need for further action. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF03_CSA N289.5_4.1.1.3_16
Document ID	CSA N289.5
Article/Clause	4.1.1.3
Requirement Assessed	Existing nuclear power plants and on-site nuclear facilities requires the seismic instrumentation to operate for the life of the plant, including outages, and recommends that a review of its capability be done every 10 years. It requires at least one free-field triaxial accelerometer, with annunciation to indicate the occurrence of any seismic event, loss of power to the system, and malfunction of the system. Instruments are required to be verified to be suitable for use at their selected location. A number of recommendations (i.e., should statements) are made about the location and number of instruments in single or multi-unit plants.
Macro-Gap	SF03-04-16
Issue/Gap Description	A free field accelerometer should be provided on the site and accelerometers should be placed on SSCs as needed to confirm that a seismic event has occurred and that the plant is operable after the event
Rationale	The off-site monitoring is an acceptable method per Clause 6.5.2 (c) of N289.1-08 which places the responsibility on the nuclear operator that the means are in place to declare a seismic event through “on-site instrument records, off site seismic data or rapid notifications from appropriate agencies such as the Geological Survey of Canada”. Procedure DPT-PDE-00017, Bruce Power Seismic Qualification Standard includes CSA N289.5 as a basis for seismic qualification (clause 4.1, second paragraph), but notes in Section 4.6 (Post Seismic Response) that notification of an earthquake of magnitude 5 or greater within 500 km of the site will be received from the Southern Ontario Seismograph Network, which has one monitoring station within 20 km of the Bruce site. This is also included in the operating procedures and has been accepted by the CNSC through the acceptance of the procedure noted above, which documents this monitoring approach. The notifications from the Geological Survey of Canada (GSC) are based on data from the National Teleseismic Network supplemented by the Southern Ontario Seismic Network. The GSC data are expert reviewed at GSC Ottawa before being reported. The GSC reporting is first an oral report within 24h of the seismic event if the seismic event exceeds the maximum acceleration for the Design Basis earthquake, otherwise within a business day, followed by written event notification. In summary, Bruce Power complies with the intent of Clause 6.5(and thus with CSA N289.5 Clause 4.1.1.3) by means of off-site monitoring, which is an acceptable alternative accepted by the CNSC. In this context, gaps associated with seismic instrumentation are specific to address the data collected using the monitoring instruments installed at different floor levels and hence not applicable based on the current methodology used by Bruce Power. Since the post-seismic event notification to the operating staff is considered to be adequate and has been accepted by the CNSC, it

	is judged that the free field motion accelerometer and placing accelerometers on structures and equipment is not essential. However, installation of in-plant monitoring equipment and developing the in-house expertise to interpret and assess earthquake data will be reviewed to establish the feasibility of implementing an in-plant monitoring network and make recommendations in alignment with the progress of the major station refurbishments made under the MCR project. Although the reporting requirements are listed in BP-PROC-00059 Event Response and Reporting, the post-seismic reporting and recording requirements will also be included in DPT-PDE-00017 and NK29-AIM-03600.1-25 as part of periodic updates of these procedures. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.1 Identification of events
Requirement Assessed	The licensee shall use a systematic process to identify events, event sequences, and event combinations ("events" hereafter in this document) that can potentially challenge the safety or control functions of the NPP. The licensee shall also identify events that may lead to fission product releases, including those related to spent fuel pools (also called irradiated fuel bays) and fuel-handling systems. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design. The identification of events will include at-power and shutdown states. The deterministic analysis should also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown analysis. Common-cause events affecting multiple reactor units on a site shall be considered. The list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. In addition to events that could challenge the safety or control functions of the NPP, safety analysis shall be performed for normal operation. Guidance The safety analysis is performed for a set of events that could lead to challenges related to the NPP's safety or control functions. These include events caused by SSC failures or human error as well as human-induced or natural common-cause events. The events considered in safety analysis could be single PIEs, sequences of several consequential events, or combinations of independent events. The set of events to be considered in safety analysis is identified using a systematic process and by taking into account: • reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams • lists of events developed for safety analysis of other NPPs, as applicable • analysis of operating experience data for similar plants • any events prescribed for inclusion in safety analysis by regulatory requirements (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants) • equipment failures, human errors and common-cause events

	identified iteratively with PSA • a cut-off frequency for common-cause events that is consistent across all events The list of identified events should be iteratively reviewed for accuracy and completeness as the plant design and safety analyses proceed. Reviews should also be periodically conducted throughout the NPP lifecycle, to account for new information and requirements. This regulatory document requires that, when identifying events, all permissible plant operating modes be considered. All operating modes used for extended periods of time should be analyzed. Modes that occur transiently or briefly can be addressed without a specific analysis, as long as it can be shown that existing safety analyses bound the behaviour and consequences of those states. NPP operating modes include, but are not limited to: • initial approach to reactor criticality • reactor start-up from shutdown through criticality to power • steady-state power operation, including both full and low power • changes in the reactor power level, including load follow modes (if employed) • reactor shutting down from power operation • shutdown in a hot standby mode • shutdown in a cold shutdown mode • shutdown in a refuelling mode or maintenance mode that opens major closures in the reactor coolant pressure boundary • shutdown in other modes or plant configurations with unique temperature, pressure or coolant inventory conditions • operation of limited duration, with some systems important to safety being unavailable For events identified by the systematic process used for this purpose, a full range of configurations and operating modes of equipment should be considered in the deterministic safety analysis. Special plant configurations may occur during major plant modifications such as plant refurbishment, lay-up, or decommissioning. These configurations should be considered, and potential events should be identified and included in the deterministic safety analysis.
Macro-Gap	SF05-02-15
Issue/Gap Description	Events initiated as a result of human errors are not explicitly identified in the Safety Report. PRA Initiating event frequency include implicitly any relevant operator error that may cause the initiating event (Gap 2).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or

	oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.1 Identification of events
Requirement Assessed	The licensee shall use a systematic process to identify events, event sequences, and event combinations ("events" hereafter in this document) that can potentially challenge the safety or control functions of the NPP. The licensee shall also identify events that may lead to fission product releases, including those related to spent fuel pools (also called irradiated fuel bays) and fuel-handling systems. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design. The identification of events will include at-power and shutdown states. The deterministic analysis should also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown analysis. Common-cause events affecting multiple reactor units on a site shall be considered. The list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. In addition to events that could challenge the safety or control functions of the NPP, safety analysis shall be performed for normal operation. Guidance The safety analysis is performed for a set of events that could lead to challenges related to the NPP's safety or control functions. These include events caused by SSC failures or human error as well as human-induced or natural common-cause events. The events considered in safety analysis could be single PIEs, sequences of several consequential events, or combinations of independent events. The set of events to be considered in safety analysis is identified using a systematic process and by taking into account: • reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams • lists of events developed for safety analysis of other NPPs, as applicable • analysis of operating experience data for similar plants • any events prescribed for inclusion in safety analysis by regulatory requirements (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants) • equipment failures, human errors and common-cause events

	identified iteratively with PSA • a cut-off frequency for common-cause events that is consistent across all events The list of identified events should be iteratively reviewed for accuracy and completeness as the plant design and safety analyses proceed. Reviews should also be periodically conducted throughout the NPP lifecycle, to account for new information and requirements. This regulatory document requires that, when identifying events, all permissible plant operating modes be considered. All operating modes used for extended periods of time should be analyzed. Modes that occur transiently or briefly can be addressed without a specific analysis, as long as it can be shown that existing safety analyses bound the behaviour and consequences of those states. NPP operating modes include, but are not limited to: • initial approach to reactor criticality • reactor start-up from shutdown through criticality to power • steady-state power operation, including both full and low power • changes in the reactor power level, including load follow modes (if employed) • reactor shutting down from power operation • shutdown in a hot standby mode • shutdown in a cold shutdown mode • shutdown in a refuelling mode or maintenance mode that opens major closures in the reactor coolant pressure boundary • shutdown in other modes or plant configurations with unique temperature, pressure or coolant inventory conditions • operation of limited duration, with some systems important to safety being unavailable For events identified by the systematic process used for this purpose, a full range of configurations and operating modes of equipment should be considered in the deterministic safety analysis. Special plant configurations may occur during major plant modifications such as plant refurbishment, lay-up, or decommissioning. These configurations should be considered, and potential events should be identified and included in the deterministic safety analysis.
Macro-Gap	SF05-02-15
Issue/Gap Description	The specified elements to be considered in event identification or all applicable modes of operation are not comprehensively covered (Gap 3).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action

	Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.1 Identification of events
Requirement Assessed	The licensee shall use a systematic process to identify events, event sequences, and event combinations (“events” hereafter in this document) that can potentially challenge the safety or control functions of the NPP. The licensee shall also identify events that may lead to fission product releases, including those related to spent fuel pools (also called irradiated fuel bays) and fuel-handling systems. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design. The identification of events will include at-power and shutdown states. The deterministic analysis should also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown analysis. Common-cause events affecting multiple reactor units on a site shall be considered. The list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. In addition to events that could challenge the safety or control functions of the NPP, safety analysis shall be performed for normal operation. Guidance The safety analysis is performed for a set of events that could lead to challenges related to the NPP’s safety or control functions. These include events caused by SSC failures or human error as well as human-induced or natural common-cause events. The events considered in safety analysis could be single PIEs, sequences of several consequential events, or combinations of independent events. The set of events to be considered in safety analysis is identified using a systematic process and by taking into account: • reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams • lists of events developed for safety analysis of other NPPs, as applicable • analysis of operating experience data for similar plants • any events prescribed for inclusion in safety analysis by regulatory requirements (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants) • equipment failures, human errors and common-cause events

	identified iteratively with PSA • a cut-off frequency for common-cause events that is consistent across all events The list of identified events should be iteratively reviewed for accuracy and completeness as the plant design and safety analyses proceed. Reviews should also be periodically conducted throughout the NPP lifecycle, to account for new information and requirements. This regulatory document requires that, when identifying events, all permissible plant operating modes be considered. All operating modes used for extended periods of time should be analyzed. Modes that occur transiently or briefly can be addressed without a specific analysis, as long as it can be shown that existing safety analyses bound the behaviour and consequences of those states. NPP operating modes include, but are not limited to: • initial approach to reactor criticality • reactor start-up from shutdown through criticality to power • steady-state power operation, including both full and low power • changes in the reactor power level, including load follow modes (if employed) • reactor shutting down from power operation • shutdown in a hot standby mode • shutdown in a cold shutdown mode • shutdown in a refuelling mode or maintenance mode that opens major closures in the reactor coolant pressure boundary • shutdown in other modes or plant configurations with unique temperature, pressure or coolant inventory conditions • operation of limited duration, with some systems important to safety being unavailable For events identified by the systematic process used for this purpose, a full range of configurations and operating modes of equipment should be considered in the deterministic safety analysis. Special plant configurations may occur during major plant modifications such as plant refurbishment, lay-up, or decommissioning. These configurations should be considered, and potential events should be identified and included in the deterministic safety analysis.
Macro-Gap	SF05-02-16
Issue/Gap Description	Events initiated as a result of human errors are not explicitly identified in the Safety Report.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action

	Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.1 Identification of events
Requirement Assessed	The licensee shall use a systematic process to identify events, event sequences, and event combinations ("events" hereafter in this document) that can potentially challenge the safety or control functions of the NPP. The licensee shall also identify events that may lead to fission product releases, including those related to spent fuel pools (also called irradiated fuel bays) and fuel-handling systems. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design. The identification of events will include at-power and shutdown states. The deterministic analysis should also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown analysis. Common-cause events affecting multiple reactor units on a site shall be considered. The list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. In addition to events that could challenge the safety or control functions of the NPP, safety analysis shall be performed for normal operation. Guidance The safety analysis is performed for a set of events that could lead to challenges related to the NPP's safety or control functions. These include events caused by SSC failures or human error as well as human-induced or natural common-cause events. The events considered in safety analysis could be single PIEs, sequences of several consequential events, or combinations of independent events. The set of events to be considered in safety analysis is identified using a systematic process and by taking into account: • reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams • lists of events developed for safety analysis of other NPPs, as applicable • analysis of operating experience data for similar plants • any events prescribed for inclusion in safety analysis by regulatory requirements (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants) • equipment failures, human errors and common-cause events

	identified iteratively with PSA • a cut-off frequency for common-cause events that is consistent across all events The list of identified events should be iteratively reviewed for accuracy and completeness as the plant design and safety analyses proceed. Reviews should also be periodically conducted throughout the NPP lifecycle, to account for new information and requirements. This regulatory document requires that, when identifying events, all permissible plant operating modes be considered. All operating modes used for extended periods of time should be analyzed. Modes that occur transiently or briefly can be addressed without a specific analysis, as long as it can be shown that existing safety analyses bound the behaviour and consequences of those states. NPP operating modes include, but are not limited to: • initial approach to reactor criticality • reactor start-up from shutdown through criticality to power • steady-state power operation, including both full and low power • changes in the reactor power level, including load follow modes (if employed) • reactor shutting down from power operation • shutdown in a hot standby mode • shutdown in a cold shutdown mode • shutdown in a refuelling mode or maintenance mode that opens major closures in the reactor coolant pressure boundary • shutdown in other modes or plant configurations with unique temperature, pressure or coolant inventory conditions • operation of limited duration, with some systems important to safety being unavailable For events identified by the systematic process used for this purpose, a full range of configurations and operating modes of equipment should be considered in the deterministic safety analysis. Special plant configurations may occur during major plant modifications such as plant refurbishment, lay-up, or decommissioning. These configurations should be considered, and potential events should be identified and included in the deterministic safety analysis.
Macro-Gap	SF05-02-16
Issue/Gap Description	The specified elements to be considered in event identification and plant operating modes are not comprehensively covered.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action

	Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.2.5_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.2.5 Guidance for combinations of events
Requirement Assessed	Combinations of events (which may occur either simultaneously or sequentially while restoring the plant to a stable state) should be considered. Types of combinations include: • multiple independent failures in equipment important to safety • failure of a process system and system important to safety • multiple process system failures • equipment failures and operator errors • common-cause events and operator errors Examples of event combinations include: • loss of coolant with subsequent loss of station electrical power, including station blackout • loss of coolant with loss of containment cooling • small loss-of-coolant accidents (LOCAs) with failure of primary or secondary depressurization • main steam line break with failure of the operator to initiate a backup cooling system
Macro-Gap	SF05-02-16
Issue/Gap Description	Not all types of event combinations indicated in guidance clause 4.2.2.5 have been considered. For example, common-cause events and operator errors are not considered.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-02-16
Issue/Gap Description	Provisions for DECs were not explicitly considered in the design basis
Rationale	At this time the SRI Project is already introducing the following elements as related to BDBAs and hence DECs: -Screening of events to identify legacy design basis accident events which more appropriately are to be classified as BDBAs and the movement of their existing SR content to a new Appendix 12 on BDBAs, and; -Addition of BDBA Summaries, as required; As shown in Figure 1 of REGDOC-2.4.1, Design Extension Conditions (DEC) are a subset of BDBAs which are characterized by 'no severe fuel degradation' or 'Severe Accidents'. A severe accident is defined as 'An accident more severe than a design-basis accident and involving severe fuel degradation in the reactor core or spent fuel pool.' However, there is no quantitative criteria is available to characterize and differentiate fuel degradation in terms of severity such that the set of accidents that are to be addressed as DECs can be established. It is recognized that a common industry position and an alternative technical guidance is essential for addressing DECs in DSA. Currently neither has been established. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-03-15
Issue/Gap Description	Dose calculation of Part 3 of the Safety Report are not completely consistent with the guidance in Section 4.4.4.7 (Gap 2).

Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-01-15
Issue/Gap Description	Guidance on specific assumptions related to crediting passive functions of containment system in AOOs dose calculation cannot be assessed since

	AOOs are not separately analyzed in Part 3 of the Safety Report (Gap 3).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-03-15
Issue/Gap Description	Quantitative acceptance criteria of Part 3 of the Safety Report are based on direct physical evidence and well-understood phenomena, but

	accounting for uncertainties is not demonstrated (Gap 4).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-01-15
Issue/Gap Description	Compliance with the guidance on qualitative acceptance criteria for AOOs cannot be assessed since the analysis in Part 3 of the Safety Report does

	not consider AOOs separately. However, it is expected that most of the guidance elements can be demonstrated (Gap 5).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-02-16
Issue/Gap Description	Guidance on specific assumptions related to crediting passive functions of the containment system in AOOs dose calculation cannot be assessed

	since AOOs are not separately analyzed in Part 3 of the Safety Report (Gap 3).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-03-16
Issue/Gap Description	Quantitative acceptance criteria of Part 3 of the Safety Report are based on well-understood phenomena, but are not systematically supported by

	experimental data (Gap 4).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-02-16
Issue/Gap Description	Compliance with the guidance on qualitative acceptance criteria for AOOs cannot be assessed since the analysis in Part 3 of the Safety Report does

	not consider AOOs separately. However, it is expected that most of the guidance elements can be demonstrated (Gap 5).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-06-16
Issue/Gap Description	Dose calculations of Part 3 of the Safety Report are not completely consistent with the guidance in Section 4.4.4.7 (Gap 2).

Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-15
Issue/Gap Description	Incorporation of margins or Safety Factors to account for uncertainty in experimental data and relevant models has not been systematically demonstrated in selecting the quantitative acceptance criteria (Gap 4).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-15
Issue/Gap Description	A more conservative quantitative acceptance criterion has not been selected in some cases where qualified models with high confidence does not exist (e.g. For events with high fuel sheath temperatures exceeding 1500 C (Gap 3).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-16
Issue/Gap Description	Incorporation of margins or safety factors to account for uncertainty in experimental data and relevant models has not been systematically demonstrated in selecting the quantitative acceptance criteria (Gap 4).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-16
Issue/Gap Description	A more conservative quantitative acceptance criterion has not been selected in some cases where qualified models with high confidence does not exist (e.g. For events with high fuel sheath temperatures exceeding 1500 C).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2.9_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2.9 Guidance for documentation of results
Requirement Assessed	Results of deterministic safety analysis calculations are documented in such a way as to facilitate their review and understanding. The documentation of safety analysis results should include: • objective of the analysis • analysis assumptions and their justification • plant models and modelling assumptions • any computer code user options that differ from the options used in code validation • analysis results in comparison with acceptance criteria • findings and conclusions from sensitivity and uncertainty analyses Further guidance is provided in section 4.5.
Macro-Gap	SF05-01-15
Issue/Gap Description	The analysis in Part 3 of the Safety Report does not identify computer code user options that differ from the options used in code validation (Gap 1). DPT-NSAS-00015 on Execution of Safety Analysis address this issue requiring to stipulate that when multiple changes in code versions and/or models have occurred from a reference analysis, sensitivity studies shall be performed to determine the impact of each change on the specific analysis.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2.9_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2.9 Guidance for documentation of results
Requirement Assessed	Results of deterministic safety analysis calculations are documented in such a way as to facilitate their review and understanding. The documentation of safety analysis results should include: • objective of the analysis • analysis assumptions and their justification • plant models and modelling assumptions • any computer code user options that differ from the options used in code validation • analysis results in comparison with acceptance criteria • findings and conclusions from sensitivity and uncertainty analyses Further guidance is provided in section 4.5.
Macro-Gap	SF05-01-16
Issue/Gap Description	The analysis in Part 3 of the Safety Report does not identify computer code user options that differ from the options used in code validation.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4.6_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4.6 Guidance for modelling assumptions
Requirement Assessed	The assumptions incorporated in the computer codes, or made during code applications, should be such that safety analysis results (whether best-estimate or conservative) remain physically sound. In performing safety analysis, justifications should be provided for all instances where the assumptions used are different than those used in the validation.
Macro-Gap	SF05-01-15
Issue/Gap Description	Safety Report analyses do not include assessment whether code model options used in the analysis are similar to those used in their validation (Gap 1).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4.6_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4.6 Guidance for modelling assumptions
Requirement Assessed	The assumptions incorporated in the computer codes, or made during code applications, should be such that safety analysis results (whether best-estimate or conservative) remain physically sound. In performing safety analysis, justifications should be provided for all instances where the assumptions used are different than those used in the validation.
Macro-Gap	SF05-01-16
Issue/Gap Description	Safety Report analyses do not include an assessment of whether code model options used in the analysis are similar to those used in their validation.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4.7_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4.7 Guidance for dose calculations
Requirement Assessed	As mentioned in section 4.3, the committed whole-body dose for average members of the critical groups who are most at risk (at or beyond the site boundary) is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. The effective dose should be used in dose calculations, and should include contributions from: • external radiation from cloud and ground deposits • inhaled radioactive materials • skin absorption of tritium In dose calculations, the worst weather scenario in terms of predicted dose should be assumed. All weather scenarios with probabilities of occurrences higher than 5 percent should be accounted for. No intervention in the form of decontamination or evacuation should be assumed. Intervention against ingestion of radioactive materials and natural removal processes may be assumed. Dose calculations should also be conducted for several time intervals, and up to one year after the accident.
Macro-Gap	SF05-07-15
Issue/Gap Description	Part 3 of the Safety Report does not demonstrate whether it covers weather scenarios with probabilities of occurrences higher than 5% and does not include dose calculations for intervals up to 1 year (Gap1).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4.7_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4.7 Guidance for dose calculations
Requirement Assessed	As mentioned in section 4.3, the committed whole-body dose for average members of the critical groups who are most at risk (at or beyond the site boundary) is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. The effective dose should be used in dose calculations, and should include contributions from: • external radiation from cloud and ground deposits • inhaled radioactive materials • skin absorption of tritium In dose calculations, the worst weather scenario in terms of predicted dose should be assumed. All weather scenarios with probabilities of occurrences higher than 5 percent should be accounted for. No intervention in the form of decontamination or evacuation should be assumed. Intervention against ingestion of radioactive materials and natural removal processes may be assumed. Dose calculations should also be conducted for several time intervals, and up to one year after the accident.
Macro-Gap	SF05-06-16
Issue/Gap Description	Part 3 of the Safety Report does not demonstrate whether it covers weather scenarios with probabilities of occurrences higher than 5% and does not include dose calculations for intervals up to 1 year.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.6_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.6 Conservatism in deterministic safety analysis
Requirement Assessed	The safety analysis shall build in a degree of conservatism to off-set any uncertainties associated with both NPP initial and boundary conditions and modelling of NPP performance in the analyzed event. This conservatism shall depend on event class and shall be commensurate with the analysis objectives. Guidance Safety analysis needs to incorporate a degree of conservatism that is commensurate with the safety analysis objectives and is dependent on the event class. Conservatism in safety analysis is often necessary to cover the potential impact of uncertainties, and may be achieved through judicious application of conservative assumptions and data. The concept of conservatism is applied to Level 3 defence-in-depth safety analysis. This is to ensure that limiting assumptions are used when knowledge of the physical phenomena is insufficient. For Level 2 and Level 4 defence in depth, the safety analysis should be carried out using best-estimate assumptions, data and methods. Where this is not possible, a reasonable degree of conservatism (appropriate for the objectives of these levels) should be used, to compensate for the lack of adequate knowledge concerning the physical processes governing these events. While it is permissible – and sometimes encouraged – to use conservative codes, it is usually preferable to apply realistic (best-estimate) computer codes. Where conservative analysis results are required for Level 3 defence-in-depth (AOO and DBA) analysis, best-estimate computer codes should be used along with the assessment of modelling and input plant parameter uncertainties. The deterministic safety analysis for AOO and DBA (conservative analysis for Level 3 defence in depth) should: • apply the single-failure criterion to all safety groups, and ensure that the safety groups are environmentally and seismically qualified • use minimum allowable performance (as established in the OLCs) for safety groups • account for consequential failures that may occur as a result of the initiating event • credit the actions of process and control systems only where the systems are passive and environmentally and seismically qualified for the accident conditions

	• include the actions of process and control systems when their actions may have a detrimental effect on the consequences of the analyzed accident • credit the normally running process systems that are not affected by the analyzed accident • if operator actions are credited, demonstrate that credible "worst case" operator performance has been considered in the analysis and assessment Independent selection of all parameters at their conservative values can lead to plant states that are not physically feasible. When this could be the case, it is recommended to select conservatively those key parameters that have the strongest influence on the results in comparison with the acceptance criterion under consideration. The remaining parameters can be specified more consistently in the ensuing calculations. Each calculation should account for the impact of a particular parameter, so that the effects of all parameters can be assessed.
Macro-Gap	SF05-09-15
Issue/Gap Description	Conservative assumptions are used in the analysis. However, there is no demonstration that the conservatism of the analysis would cover modeling uncertainties (Gap 1).
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.6_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.6 Conservatism in deterministic safety analysis
Requirement Assessed	The safety analysis shall build in a degree of conservatism to off-set any uncertainties associated with both NPP initial and boundary conditions and modelling of NPP performance in the analyzed event. This conservatism shall depend on event class and shall be commensurate with the analysis objectives. Guidance Safety analysis needs to incorporate a degree of conservatism that is commensurate with the safety analysis objectives and is dependent on the event class. Conservatism in safety analysis is often necessary to cover the potential impact of uncertainties, and may be achieved through judicious application of conservative assumptions and data. The concept of conservatism is applied to Level 3 defence-in-depth safety analysis. This is to ensure that limiting assumptions are used when knowledge of the physical phenomena is insufficient. For Level 2 and Level 4 defence in depth, the safety analysis should be carried out using best-estimate assumptions, data and methods. Where this is not possible, a reasonable degree of conservatism (appropriate for the objectives of these levels) should be used, to compensate for the lack of adequate knowledge concerning the physical processes governing these events. While it is permissible – and sometimes encouraged – to use conservative codes, it is usually preferable to apply realistic (best-estimate) computer codes. Where conservative analysis results are required for Level 3 defence-in-depth (AOO and DBA) analysis, best-estimate computer codes should be used along with the assessment of modelling and input plant parameter uncertainties. The deterministic safety analysis for AOO and DBA (conservative analysis for Level 3 defence in depth) should: • apply the single-failure criterion to all safety groups, and ensure that the safety groups are environmentally and seismically qualified • use minimum allowable performance (as established in the OLCs) for safety groups • account for consequential failures that may occur as a result of the initiating event • credit the actions of process and control systems only where the systems are passive and environmentally and seismically qualified for the accident conditions

	• include the actions of process and control systems when their actions may have a detrimental effect on the consequences of the analyzed accident • credit the normally running process systems that are not affected by the analyzed accident • if operator actions are credited, demonstrate that credible “worst case” operator performance has been considered in the analysis and assessment Independent selection of all parameters at their conservative values can lead to plant states that are not physically feasible. When this could be the case, it is recommended to select conservatively those key parameters that have the strongest influence on the results in comparison with the acceptance criterion under consideration. The remaining parameters can be specified more consistently in the ensuing calculations. Each calculation should account for the impact of a particular parameter, so that the effects of all parameters can be assessed.
Macro-Gap	SF05-07-16
Issue/Gap Description	There is no demonstration that the conservatism of the analysis would cover modeling uncertainties.
Rationale	Gaps related to the guidance sections of REGDOC-2.4.1 do not have the same safety or regulatory significance as gaps in the paragraphs that describe the requirements. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT 2015_4.1_15
Document ID	SF8 RT 2015
Article/Clause	4.1
Requirement Assessed	Key Implementing Documents The key Bruce Power documents related to implementation of the elements related to Safety Performance are indicated in Table 4 of SF 8 Report.
Macro-Gap	SF08-10-15
Issue/Gap Description	BP-PROC-00136 Plant Operational Review Committee (PORC) and BP-PROC-00169 Safety Related System List are not affiliated with a Program
Rationale	BP-PROC-00136 is now explicitly mentioned in BP-PROG-12.01 Conduct of Plant Operations in Sections 4.2.3.30 and 5.3 and in Appendices B and C. BP-PROC-00169 states that procedure takes authority from BP-MSM-1, Management System Manual, and BP-PROG-03.01, Document Management. However, this is not explicitly noted as such in BP-MSM-1 or BP-PROG-03.01. This gap is categorized as “Closed” in the PSR database.

Gap #	SF08_SF8 RT 2015_5.13_15
Document ID	SF8 RT 2015
Article/Clause	5.13
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: k) Compliance with regulatory requirements
Macro-Gap	SF08-07-15
Issue/Gap Description	Produce a document that explains the relationship and impact of the Fukushima type changes on the design basis, safety analyses and assessments, as they have been included in the licensing basis. This is necessary to ensure that the Design Basis and Configuration Management implications are understood. As appropriate, ensure Design Requirement and Design Manuals are updated appropriately, including capturing of Design Extension conditions if appropriate.
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT 2015_5.13_15
Document ID	SF8 RT 2015
Article/Clause	5.13
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: k) Compliance with regulatory requirements
Macro-Gap	SF08-01-15
Issue/Gap Description	Governance procedures for the Integrated or Periodic Safety Review process need to be finalized to ensure staff understanding of the Regulatory direction.
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT 2015_5.14.2_15
Document ID	SF8 RT 2015
Article/Clause	5.14.2
Requirement Assessed	The review considers the effectiveness of the processes and methodology used to evaluate and assess operating experience and trends. The findings of the reviews of other Safety Factors is taken into account when undertaking this task.
Macro-Gap	SF08-02-15
Issue/Gap Description	A risk-informed decision making process should be included in Equipment Reliability program so as to continually better prioritize activities.
Rationale	This is a process effectiveness improvement gap which can be implemented through Bruce Power's current governance. It should be noted that Equipment Reliability program employs risk insights in terms of establishing the scope of SSCs covered under the program and uses risk significance measures in establishing surveillance, testing and inspection activities. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT 2015_5.7_15
Document ID	SF8 RT 2015
Article/Clause	5.7
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: e) Modifications, either temporary or permanent, to SSCs important to safety
Macro-Gap	SF08-04-15
Issue/Gap Description	The integrated time frame from conceptual design to station implementation for Nuclear Safety improvements that restore or improve margins (e.g., New Neutron Trip Project) needs to be reduced.
Rationale	This is a process effectiveness improvement gap which can be implemented through Bruce Power's current governance. It should also be noted that implementation of safety improvements are based on their safety significance and regulatory commitments. Progress updates are provided as agreed with the CNSC. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_4.1_16
Document ID	SF8 RT
Article/Clause	4.1
Requirement Assessed	Key Implementing Documents
Macro-Gap	SF08-09-16
Issue/Gap Description	BP-PROC-00169 Safety Related System List is not affiliated with a Program.
Rationale	BP-PROC-00169 states that procedure takes authority from BP-MSM-1, Management System Manual, and BP-PROG-03.01, Document Management. However, this is not explicitly noted as such in BP-MSM-1 or BP-PROG-03.01. This gap is categorized as “Closed” in the PSR database.

Gap #	SF08_SF8 RT_4.7_16
Document ID	SF8 RT
Article/Clause	4.7
Requirement Assessed	Not all Bruce Power Programs readily map to the Safety Factor Reports. BP-PROC-01024 [4] should consider mapping each program to the respective Safety Factor Reports in Section 4.6 of the procedure to ensure completeness of items impacting the four pillars of safety. BP-PROC-00936 [82] should interface with BP PROC-01024 [4] as the PSR is an input to the procedure.
Macro-Gap	SF08-07-16
Issue/Gap Description	Most but not all Bruce Power Programs readily map to a Safety Factor Report. For clarity, completeness and understanding, a mapping of each program relevant to reactor safety to the respective Safety Factor Reports (e.g., in Section 4.6 of the procedure) should be considered. Examples of programs loosely affiliated with BP-PROC-01024 are: BP PROG-12.03 Fuel Management [173]; BP-PROG-05.01, Supply Chain [186]; and BP-PROG-12.02, Chemistry Management [170]. Fuel Management is briefly covered in Safety Factor Report 1 as part of the code clause-by-clause comparison for a Regulatory Document for new Nuclear Power Plants and in Safety Factor 2 as part of condition assessment. Similarly, BP-PROG-12.02 Chemistry Management is identified in Safety Factor Reports 1, 2 and 4, but no in-depth discussion is provided. Every SFR uses the Nuclear Oversight and Regulatory Affairs (NORA) Audit and FASA processes but IAEA SSG-25 (and thus, the Bruce B PSR Basis Document [5]) does not include a review of the safety importance of the NORA oversight role.
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance if necessary. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_5.13_16
Document ID	SF8 RT
Article/Clause	5.13
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: k) Compliance with regulatory requirements
Macro-Gap	SF08-06-16
Issue/Gap Description	It would assist staff in future modifications and licensing assessments if design documentation clearly explains the relationship and impact of the licensing driven changes on the design basis, safety analyses and assessments. Although a review was done against WANO SOER 2013-2 ([281] Enclosure 1, Section 2.0, footnote 3), it is unclear how the Design Basis Assumptions were reviewed and updated in the design documentation. The review shows what changes were made from a detailed design and operational perspective, but does not identify how the design guides or design requirements were changed, particularly the guides covering the nuclear safety philosophy. For example, this would help ensure that the Safety Design Guides [261] and Design Requirements/Manuals are systematically revised to incorporate the Fukushima type design changes. Deviations from the Design Guides and changes were provided to the Regulator who raised an Action Item if they did not agree with the rationale [282].
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_5.14.2_16
Document ID	SF8 RT
Article/Clause	5.14.2
Requirement Assessed	The review considers the effectiveness of the processes and methodology used to evaluate and assess operating experience and trends. The findings of the reviews of other Safety Factors is taken into account when undertaking this task.
Macro-Gap	SF08-01-16
Issue/Gap Description	As a follow-up to the audit under Action Item 2014-07-4687 - BRPD-AB-2014-002 - Condition Assessment Inspection in Section 7.3.1, BP-PROC-00498 [144] Bruce Power was reviewing this procedure for continuing applicability given the revised Ageing Management governance that Bruce Power was implementing. Bruce Power stated it is to be revised, superseded or cancelled and the process requirements defined in BP-PROC-00166 [287] is to be applied to the resulting product. Afterwards CNSC staff noted some continuing deficiencies with this condition assessment procedure, BP-PROC-00498 [144]. Bruce Power reaffirmed this procedure is to be incorporated into the aging management suite of procedures under the Equipment Reliability Program, BP-PROG-11.01 [51] and the items identified during the inspection will be considered during these revisions [288][235]
Rationale	Bruce Power reaffirmed that this procedure is to be incorporated into the aging management suite of procedures under the Equipment Reliability Program, BP-PROG-11.01 and the items identified during the inspection will be considered during these revisions per references: NK21-CORR-00531-11913 / NK29-CORR-00531-12294, Action Item 2014-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2104-002, K. Lafrenière to F. Saunders, CNSC Letter, February 4, 2015 NK29-CORR-00531-12570 / NK21-CORR-00531-12206, Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002, Bruce Power Letter, F. Saunders to K. Lafrenière, September 16, 2015 This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002 and its associated ARs.

Gap #	SF08_SF8 RT_5.3_16
Document ID	SF8 RT
Article/Clause	5.3
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: a) Safety related incidents, low level events and near misses
Macro-Gap	SF08-12-16
Issue/Gap Description	An Action Request 28508028 was raised to address a potential gap between the Safety Analysis – Analysis of Record and the allowable operating states. Bruce Power extended this AR review to confirm the links between the safety analysis and operations documents was comprehensively captured in the safe operating envelope as documented by the set of limits and allowable operating configurations in the OSRs, including the DCRs raised against these documents. This review of whether there were unidentified gaps between the safety analysis and OSRs published since 2005 is to be completed in 2016. The high priority OSRs include: Fuel and Physics, Shutdown Systems, Heat Transport System, Moderator System, Containment System and Emergency Coolant Injection System. These OSRs are to be reviewed as they constitute the majority of the updated analysis. Others like the Negative Pressure Containment System are to be reviewed after the high priority ones. A Safety Analysis Mapping Results Spreadsheet will be produced to capture all the safety analysis produced since the OSRs were issued. It will capture the changes to various parameters and limits as new analysis was produced which superseded the earlier analysis. This will then be compared to the parameters and limits in OSRs to ensure completeness and identify gaps and whether they can be fully dispositioned or make recommendations for improvement [225].
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_5.3_16
Document ID	SF8 RT
Article/Clause	5.3
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: a) Safety related incidents, low level events and near misses
Macro-Gap	SF08-02-16
Issue/Gap Description	The CNSC has requested clarification on the proposed path forward to close the gaps they have identified and meetings to discuss the remaining CANDU safety issues impacting LBLOCA [221] [222]. For completeness, this issue is flagged as a gap as the Safety Report Improvement Project [223] will need to capture changing LBLOCA analysis in future Safety Report updates as part of the Safety Report Framework update and the Safety Analysis Improvement Program [224] after the delivery of the 2017 update of the Safety Report
Rationale	This is already in place : Letter, F. Saunders to B. Howden, " CANDU Category Ill Safety Issues: Annual Update" June 16, 2016 NK21-CORR-00531-12836 / NK29-CORR-00531-1 3286 "A detailed update on the following four CSIs related to Large Break Loss of Coolant Accidents (LBLOCA) was not included in the previous annual updates because they were managed separately as part of a CANDU Owners Group (COG) Joint Project (JP#4367): AA 9 - Analysis for Void Reactivity Coefficient, PF 9 - Fuel Behaviour in High Temperature Transients, PF 10— Fuel Behaviour in Power Pulse Transients, and PF 12— GAI 00GO1 Channel Voiding during a Large LOCA As part of JP #4367. PF 12 was reclassified to Category II in Reference 3 and the LBLOCA Composite Analytical Approach was developed in support of reclassifying AA 9, PF 9 and PF 10 in Reference 4. Bruce Power has responded to CNSC staff comments and participated in workshops on specific technical areas, which has resulted in the development of a plan and schedule for a LBLOCA Composite Analytical Approach licensing application. The submission of the plan and schedule in Reference 5 is currently under review by CNSC staff, as described in Reference 6." This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_5.7_16
Document ID	SF8 RT
Article/Clause	5.7
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: e) Modifications, either temporary or permanent, to SSCs important to safety
Macro-Gap	SF08-03-16
Issue/Gap Description	The integrated time frame from conceptual design to station implementation for Nuclear Safety improvements that restore safety margins (e.g., heat transport high pressure trip on Units 3 and 4) should be reviewed to find opportunities to more efficiently implement the safety improvement.
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. It should also be noted that notifications to the CNSC are governed by BP-PROG-06.03 CNSC Interface Management. Bruce Power provides notifications of safety improvements based on their safety significance and provisions of the PROL. Progress updates are provided as agreed with the CNSC. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_7.2_16
Document ID	SF8 RT
Article/Clause	7.2
Requirement Assessed	Internal and External Audits and Reviews
Macro-Gap	SF08-10-16
Issue/Gap Description	The following PROGs, PROCs have not been revised within the required 3 year timeframe per BP PROC-00166: General Procedure and Process Requirements and a review of the PassPort action requests does not always provide evidence that the standard 3-year review has been completed and recommended no changes or whether the review has been deferred to a later date: BP-PROG-01.01-R005, Business Planning Program, February 5, 2010 [103]. BP-PROG-11.02-R006, On-Line Work Management Program, October 2012 [159]. BP-PROG-11.03-R005, Outage Work Management, July 2011 [163]. BP-PROC-00169-R002, Safety Related System List, September 2007 [182]. BP-PROC-00260-R005, Material Condition and Housekeeping, November 15, 2012 [168]. BP-PROC-00498-R006, Condition Assessment of Generating Units in Support of Life Extension, February 3, 2011 [144]. BP-PROC-00735-R002, Long Range Cycle Planning Process, August 28, 2012 [162]. BP-PROC-00795-R000, Human Performance Tools for Knowledge Workers, March 30, 2011 [102]. BP-PROC-00789-R001, Maintenance Strategy, April 23, 2014 [296]. BP-PROC-00839-R000, Reporting to CNSC/IAEA – Safeguards, June 21, 2012 [129]. DPT-NSAS-00003-R004, Guidelines for Evaluating and Prioritizing Safety Report Issues, September 2011 [134]. DPT-PE-00005-R000, Performance Requirements for Contamination Exhaust Control Filters, February 23, 2005 [148]. SEC-EQD-00035-R002, Environmental Qualification Sustainability Monitoring, November 15, 2012 [131].
Rationale	Bruce Power reaffirmed that this procedure is to be incorporated into the aging management suite of procedures under the Equipment Reliability Program, BP-PROG-11.01 and the items identified during the inspection will be considered during these revisions per references: NK21-CORR-00531-11913 / NK29-CORR-00531-12294, Action Item 2014-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2104-002, K. Lafrenière to F. Saunders, CNSC Letter, February 4, 2015 NK29-CORR-00531-12570 / NK21-CORR-00531-12206, Action Item

	2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002, Bruce Power Letter, F. Saunders to K. Lafrenière, September 16, 2015 This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002 and its associated ARs (Action Requests).

Gap #	SF08_SF8 RT_7.2_16
Document ID	SF8 RT
Article/Clause	7.2
Requirement Assessed	Internal and External Audits and Reviews
Macro-Gap	SF08-11-16
Issue/Gap Description	Audit raised ARs against the BP-PROC-00666 to change the document, but this was not flagged in PassPort
Rationale	This is a documentation update gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF08_SF8 RT_7.2_16
Document ID	SF8 RT
Article/Clause	7.2
Requirement Assessed	Internal and External Audits and Reviews
Macro-Gap	SF08-08-16
Issue/Gap Description	Update the procedures to consider lessons learned from INPO 05-008 [80]
Rationale	"This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF09_SF9 RT 2015_5.3.1_15
Document ID	SF9 RT 2015
Article/Clause	5.3.1
Requirement Assessed	Research Bruce Power is heavily invested in Research and Development to support ongoing operations. This occurs in many different areas and disciplines
Macro-Gap	SF09-01-15
Issue/Gap Description	Bruce Power participates widely in external conferences, symposia, research projects, but no specific governance was found that fosters this participation other than tangential references in BP-MSM-1 Sheet 2 and BP PROG-09.02.
Rationale	This gap was identified as part of the initial assessment provided to Bruce Power for review and comment. It was assessed as not being a gap based on the additional information provided by Bruce Power.

Gap #	SF09_SF9 RT_5.3.1.2_16
Document ID	SF9 RT
Article/Clause	5.3.1.2
Requirement Assessed	The review is conducted in accordance with the Bruce B PSR Basis Document, which states that the review tasks are as follows: 3. Review the processes for assessing and, if necessary, implementing research findings and findings from operating experience relevant to safety: 1.2. Participation in Canadian Standards Association.
Macro-Gap	SF09-01-16
Issue/Gap Description	While the cerebral transport of knowledge is implicit in the stature and qualifications of the staff appointed to the CSA committees, governance surrounding their collection and use of OPEX in performing their duties in the various committees has not been found.
Rationale	Participants in CSA standards are bound to operate per Bruce Power's and CSA's governance. Specifically, use of OPEX is governed per BP-PROG-01.06 and includes all activities associated with plant operation and operational support. This includes collection and use of OPEX by staff appointed to the CSA committees while performing their duties in the various committees. Hence, such an improvement is judged to be unnecessary. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF10_SF10 RT 2015_4.1_15
Document ID	SF10 RT 2015
Article/Clause	4.1
Requirement Assessed	Key Implementing Documents
Macro-Gap	SF10-04-15
Issue/Gap Description	BP-PROC-00136 and BP-PROC-00169 Safety Related System List are not affiliated with a Program.
Rationale	BP-PROC-00136 is now explicitly mentioned in BP PROG-12.01 Conduct of Plant Operations in Sections 4.2.3.30 and 5.3 and in Appendices B and C. BP-PROC-00169 states that procedure takes authority from BP-MSM-1, Management System Manual, and BP-PROG-03.01, Document Management. However, this is not explicitly noted as such in BP-MSM-1 or BP-PROG-03.01. This gap is categorized as “Closed” in the database.

Gap #	SF10_SF10 RT 2015_5.2.3_15
Document ID	SF10 RT 2015
Article/Clause	5.2.3
Requirement Assessed	The review of the organization and management system will include a review of the following elements or programs against national and international standards: c) The adequacy of arrangements for managing and retaining responsibility for activities or processes important to safety that have been outsourced (for example, maintenance and engineering services and safety analysis)
Macro-Gap	SF10-02-15
Issue/Gap Description	BP-PROC-00363 [77], Nuclear Safety Assessment, Section 7.1 states that the Manager of the Nuclear Safety Analysis and Support Department (NSASD) is the code owner for software and is accountable for quality, development, verification, validation, documentation, maintenance and configuration management of Nuclear Safety Analysis work, and the data sets used, and codes executed within NSASD. No discussion is explicitly provided on safety analysis produced outside of the department. Its lower tier documents DIV-ENG-00013 Planning of Internal Work for Nuclear Safety Analysis and DPT-NSAS-00008 Management of External Work for Nuclear Safety Analysis provide no guidance on the responsibility for work outside the department. BP-PROC-00363, Nuclear Safety Assessment, and its implementing documents should be revised to provide guidance on the responsibility of staff for Safety Analysis work performed outside of the NSAS Department
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF10_SF10 RT 2015_5.2.5_15
Document ID	SF10 RT 2015
Article/Clause	5.2.5
Requirement Assessed	The review of the organization and management system will include a review of the following elements or programs against national and international standards: e) The processes and supporting information that explain how work is to be specified, prepared, reviewed, performed, recorded, assessed and improved
Macro-Gap	SF10-01-15
Issue/Gap Description	Work Management Program (BP-PROG-11.03) does not address recurring outage issues identified through audits and FASAs
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF10_SF10 RT 2015_5.3.3_15
Document ID	SF10 RT 2015
Article/Clause	5.3.3
Requirement Assessed	In addition, the review of the organization and management system will verify the following: c) There is adequate control of documents, products and records and this information is readily retrievable
Macro-Gap	SF10-03-15
Issue/Gap Description	From section 5.3.3: ....A review of the DCRs showed many have been logged against a particular document, but have not progressed past the initiation phase. This daily verification shows that Bruce Power has an effective document management system. Assessments and Audits from Section 7 did not identify specific shortcomings against the controlled document, products or record processes. This was confirmed by a review of the assessments and audits, and CNSC inspections in Sections 7.1, 7.2, and 7.3. The FASAs in Table 5 and Table 6 identify past Bruce Power reviews relevant to this review task. FASA SA-BS-2012-01 specifically identifies the shortcoming that DCRs can become stagnant in the system, for example depending on how they are initiated. This occurs as a finding in other FASAs and Audits; for example, AU 2013 00015, where 18 outstanding DCRs were initiated prior to the revision date of a document, but they were not factored into the revision. DCRs can become stagnant in the system, for example, depending on how they are initiated. Improvements are needed in DCR completion rate.
Rationale	BP has implemented metrics in 2016 for measuring the average age of DCRs and total DCRs outstanding/resolved. BP has also implemented metrics measuring the 3 year review and the backlog (>3 years). If the procedural requirements of updating our governance every 3 years are met then the procedural requirements of incorporating all DCRs are met. Due to this new process the gap is considered closed as there is a process for tracking the DCR completion rate. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF10_SF10 RT_5.2.3_16
Document ID	SF10 RT
Article/Clause	5.2.3
Requirement Assessed	
Macro-Gap	SF10-01-16
Issue/Gap Description	BP-PROC-00363 [97], Nuclear Safety Assessment, Section 7.1 states that the Manager of the Nuclear Safety Analysis and Support Department (NSASD) is the code owner for safety analysis software and is accountable for quality, development, verification, validation, documentation, maintenance and configuration management of Nuclear Safety Analysis work, and the data sets used, and codes executed within NSASD. No discussion is explicitly provided on safety assessments produced outside of the department. Its lower level documents DIV-ENG-00013, Planning of Internal Work for Nuclear Safety Analysis [149] and DPT-NSAS-00008, Management of External Work for Nuclear Safety Analysis and Support [99] provide no guidance on the responsibility for work outside the department.
Rationale	This is a process improvement gap which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF10_SF10 RT_5.3.3_16
Document ID	SF10 RT
Article/Clause	5.3.3
Requirement Assessed	1. The review of the organization and management system will verify: c) There is adequate control of documents, products and records and this information is readily retrievable;
Macro-Gap	SF10-02-16
Issue/Gap Description	Many ARs Closed with a DCR which has not Been Properly Dispositioned
Rationale	BP has implemented metrics in 2016 for measuring the average age of DCRs and total DCRs outstanding/resolved. BP has also implemented metrics measuring the 3 year review and the backlog (>3 years). If the procedural requirements of updating our governance every 3 years are met then the procedural requirements of incorporating all DCRs are met. Due to this new process the gap is considered closed as there is a process for tracking the DCR completion rate. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF10_SF10 RT_5.3.3_16
Document ID	SF10 RT
Article/Clause	5.3.3
Requirement Assessed	1. The review of the organization and management system will verify: c) There is adequate control of documents, products and records and this information is readily retrievable;
Macro-Gap	SF10-03-16
Issue/Gap Description	A number of governance documents contain out of date references (e.g., superseded CNSC documents). The other issue with some of Bruce Power's governance documents is discrepancies in referencing some of the regulatory documents or standards where superseded documents are still referenced in governance documents. For instance, Regulatory Standard S-296 [137] has been replaced with CNSC REGDOC-2.9.1 [138]. However, the latest version of BP-PROG-00.02 still references S-296. There are also procedural documents that cite CSA N286-05 instead of CSA N286-12. Additionally, there are issues with out of date references in governance documents (e.g., CNSC documents that have been superseded). Some examples of these issues with various governance documents are identified in this Safety Factor Report, along with Safety Factor Reports 9, 14 and 15.
Rationale	This is a process effectiveness improvement gap which can be implemented through Bruce Power's current governance . Such discrepancies could occur as a result of the timing of updates to programs and procedures. An improvement to controlled documents procedures can minimize this type of occurrence. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF11_CSA N292.3-14_11.2.1_15
Document ID	CSA N292.3-14
Article/Clause	11.2.1
Requirement Assessed	Storage for decay should be considered to allow short-lived radionuclides to decay. Notes: 1) Decay can lower intermediate-level waste to low-level waste and permit clearance of radioactive wastes. 2) A decay storage period of 10 half-lives reduces the initial radioactivity to less than one-thousandth of its original radioactivity. 3) Storage for decay is particularly suitable for radioactive wastes containing only short-lived radionuclides. It is most suitable for wastes containing radionuclides with an approximate half-life of less than 100 d [i.e., very short-lived low-level radioactive waste (VSLLW); see Clause A.5.2 of CSA N292.0]. However, radionuclides with longer half-lives may also be considered. 4) While storage for decay can be used for bio-hazardous radioactive waste and for other perishable radioactive waste such as animal carcasses, these types of radioactive waste pose special hazards and should be segregated and stored in dedicated and monitored freezer/refrigerator cabinets during decay storage.
Macro-Gap	SF11-04-15
Issue/Gap Description	The concept of “storage for decay” is not identified in Bruce Power documentation.
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power’s radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. Radioactive waste generated at Bruce Power is not separated by isotope. Based on the waste characterization, there is a mix of long lived and short lived isotopes in the low level radioactive waste. As such, the period to store low level radioactive waste for decay cannot be minimized by segregating short lived isotopes. The long term storage of low level radioactive waste is not practicable as it would require volumes of radioactive waste to be stored for long periods of time, increasing the volume of radioactive waste being stored with the Operating station. This gap is considered to be unnecessary to implement as part of the IIP. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_11.2.1_16
Document ID	CSA N292.3-14
Article/Clause	11.2.1
Requirement Assessed	Storage for decay should be considered to allow short-lived radionuclides to decay. Notes: 1) Decay can lower intermediate-level waste to low-level waste and permit clearance of radioactive wastes. 2) A decay storage period of 10 half-lives reduces the initial radioactivity to less than one-thousandth of its original radioactivity. 3) Storage for decay is particularly suitable for radioactive wastes containing only short-lived radionuclides. It is most suitable for wastes containing radionuclides with an approximate half-life of less than 100 d [i.e., very short-lived low-level radioactive waste (VSLLW); see Clause A.5.2 of CSA N292.0]. However, radionuclides with longer half-lives may also be considered. 4) While storage for decay can be used for bio-hazardous radioactive waste and for other perishable radioactive waste such as animal carcasses, these types of radioactive waste pose special hazards and should be segregated and stored in dedicated and monitored freezer/refrigerator cabinets during decay storage.
Macro-Gap	SF11-03-16
Issue/Gap Description	The concept of “storage for decay” is not identified in Bruce Power documentation.
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power’s radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. Radioactive waste generated at Bruce Power is not separated by isotope. Based on the waste characterization, there is a mix of long lived and short lived isotopes in the low level radioactive waste. As such, the period to store low level radioactive waste for decay cannot be minimized by segregating short lived isotopes. The long term storage of low level radioactive waste is not practicable as it would require volumes of radioactive waste to be stored for long periods of time, increasing the volume of radioactive waste being stored with the Operating station. This gap is considered to be unnecessary to implement as part of the IIP. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_11.2.2_15
Document ID	CSA N292.3-14
Article/Clause	11.2.2
Requirement Assessed	The radioactive waste should a) be kept segregated from the time of generation to the end of the decay storage period; and b) have representative measurements taken, or samples taken and analyzed, prior to the removal of each batch from control (see CSA N292.5). Note: Storage for decay and clearance from further regulatory control requires strict administrative control measures.
Macro-Gap	SF11-04-15
Issue/Gap Description	The concept of “storage for decay” is not identified in Bruce Power documentation.
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power’s radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. Radioactive waste generated at Bruce Power is not separated by isotope. Based on the waste characterization, there is a mix of long lived and short lived isotopes in the low level radioactive waste. As such, the period to store low level radioactive waste for decay cannot be minimized by segregating short lived isotopes. The long term storage of low level radioactive waste is not practicable as it would require volumes of radioactive waste to be stored for long periods of time, increasing the volume of radioactive waste being stored with the Operating station. This gap is considered to be unnecessary to implement as part of the IIP. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_11.2.2_16
Document ID	CSA N292.3-14
Article/Clause	11.2.2
Requirement Assessed	The radioactive waste should a) be kept segregated from the time of generation to the end of the decay storage period; and b) have representative measurements taken, or samples taken and analyzed, prior to the removal of each batch from control (see CSA N292.5). Note: Storage for decay and clearance from further regulatory control requires strict administrative control measures.
Macro-Gap	SF11-03-16
Issue/Gap Description	The concept of “storage for decay” is not identified in Bruce Power documentation.
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power’s radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. Radioactive waste generated at Bruce Power is not separated by isotope. Based on the waste characterization, there is a mix of long lived and short lived isotopes in the low level radioactive waste. As such, the period to store low level radioactive waste for decay cannot be minimized by segregating short lived isotopes. The long term storage of low level radioactive waste is not practicable as it would require volumes of radioactive waste to be stored for long periods of time, increasing the volume of radioactive waste being stored with the Operating station. This gap is considered to be unnecessary to implement as part of the IIP. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_11.2.3_15
Document ID	CSA N292.3-14
Article/Clause	11.2.3
Requirement Assessed	Radioactive wastes containing very short-lived radionuclides (less than 100 d) should be segregated and accumulated in a separate storage area.
Macro-Gap	SF11-04-15
Issue/Gap Description	The concept of “storage for decay” is not identified in Bruce Power documentation.
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power’s radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. Radioactive waste generated at Bruce Power is not separated by isotope. Based on the waste characterization, there is a mix of long lived and short lived isotopes in the low level radioactive waste. As such, the period to store low level radioactive waste for decay cannot be minimized by segregating short lived isotopes. The long term storage of low level radioactive waste is not practicable as it would require volumes of radioactive waste to be stored for long periods of time, increasing the volume of radioactive waste being stored with the Operating station. This gap is considered to be unnecessary to implement as part of the IIP. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_11.2.3_16
Document ID	CSA N292.3-14
Article/Clause	11.2.3
Requirement Assessed	Radioactive wastes containing very short-lived radionuclides (less than 100 d) should be segregated and accumulated in a separate storage area.
Macro-Gap	SF11-03-16
Issue/Gap Description	The concept of “storage for decay” is not identified in Bruce Power documentation.
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power’s radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. Radioactive waste generated at Bruce Power is not separated by isotope. Based on the waste characterization, there is a mix of long lived and short lived isotopes in the low level radioactive waste. As such, the period to store low level radioactive waste for decay cannot be minimized by segregating short lived isotopes. The long term storage of low level radioactive waste is not practicable as it would require volumes of radioactive waste to be stored for long periods of time, increasing the volume of radioactive waste being stored with the Operating station. This gap is considered to be unnecessary to implement as part of the IIP. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_8.7_16
Document ID	CSA N292.3-14
Article/Clause	8.7 Explosivity, pyrophoricity, and chemical reactivity
Requirement Assessed	Prior to packaging, radioactive wastes shall be subject to treatments that mitigate explosivity, pyrophoricity, and chemical reactivity to the greatest extent possible.
Macro-Gap	SF11-05-16
Issue/Gap Description	Bruce Power governance documents associated with Management of Radioactive Waste do not provide any information with respect to "treatments that mitigate exclusivity, pyrophoricity, and chemical reactivity".
Rationale	Bruce Power does not treat radioactive waste prior to packaging. Bruce Power's radioactive waste is sent to OPG for long term storage, processing and disposal, packaging requirements and allowable treatments are defined by OPG in the OPG Waste acceptance Criteria. This gap is considered to be unnecessary to implement. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_9.1_15
Document ID	CSA N292.3-14
Article/Clause	9.1 General
Requirement Assessed	The selection of a radioactive waste processing method should include assessment of a) the characteristics of the radioactive wastes to be processed; b) the characteristics of the processed radioactive waste; c) the need for removal or neutralization of hazardous components; d) the waste acceptance criteria of waste management facilities which will receive the processed wastes; e) a cost/benefit analysis of radioactive waste processing as it pertains to handling, packaging, transportation, storage, and long-term management; f) the maturity of technologies in relation to minimizing processing risks; g) the risk and/or effects of radiological and conventional emissions during processing; h) ALARA in relation to facility worker exposure during handling, worker and public radiation exposure, and environmental impact risk; i) the impact of the volume reduction achieved; j) the impact of mixing long- and short-lived radionuclides and/or radioactive wastes from different points of origin; k) the availability of qualified personnel; l) the availability of other on-site processing equipment; m) transportation requirements; n) licence restrictions and regulatory requirements; and o) the complexity of and time required for regulatory approvals.
Macro-Gap	SF11-02-15
Issue/Gap Description	The selection of a radioactive waste processing method should include assessment of the maturity of technologies in relation to minimizing processing risks. This requirement is not explicitly identified in the Bruce Power procedures.
Rationale	All radioactive waste generated by Bruce Power must comply with the Waste Acceptance Criteria of OPG. As such, Bruce Power does not process radioactive waste onsite. Therefore, there is no need for guidance on selection of radioactive waste processing methods since Bruce Power does not process radioactive waste. This gap is considered to be unnecessary to implement. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_9.1_16
Document ID	CSA N292.3-14
Article/Clause	9.1 General
Requirement Assessed	The selection of a radioactive waste processing method should include assessment of a) the characteristics of the radioactive wastes to be processed; b) the characteristics of the processed radioactive waste; c) the need for removal or neutralization of hazardous components; d) the waste acceptance criteria of waste management facilities which will receive the processed wastes; e) a cost/benefit analysis of radioactive waste processing as it pertains to handling, packaging, transportation, storage, and long-term management; f) the maturity of technologies in relation to minimizing processing risks; g) the risk and/or effects of radiological and conventional emissions during processing; h) ALARA in relation to facility worker exposure during handling, worker and public radiation exposure, and environmental impact risk; i) the impact of the volume reduction achieved; j) the impact of mixing long- and short-lived radionuclides and/or radioactive wastes from different points of origin; k) the availability of qualified personnel; l) the availability of other on-site processing equipment; m) transportation requirements; n) licence restrictions and regulatory requirements; and o) the complexity of and time required for regulatory approvals.
Macro-Gap	SF11-01-16
Issue/Gap Description	The selection of a radioactive waste processing method should include assessment of the maturity of technologies in relation to minimizing processing risks. This requirement is not explicitly identified in the Bruce Power procedures.
Rationale	All radioactive waste generated by Bruce Power must comply with the Waste Acceptance Criteria of OPG. As such, Bruce Power does not process radioactive waste onsite. Therefore, there is no need for guidance on selection of radioactive waste processing methods since Bruce Power does not process radioactive waste. This gap is considered to be unnecessary to implement. This gap is categorized as "Closed" in the PSR Database.

Gap #	SF11_CSA N292.3-14_9.2.6_15
Document ID	CSA N292.3-14
Article/Clause	9.2.6 Dismantling and segmentation
Requirement Assessed	Dismantling and segmentation of equipment and/or structures should be considered to reduce radioactive waste volumes and to yield an improved packaging efficiency. Note: A variety of techniques can be used depending on factors such as the material of construction, the size and shape of the equipment, and the degree of contamination. Tools used include hand tools, saws, shears, impact tools, and cutting tools. Highly contaminated portions of the equipment and/or structures may be removed to facilitate better management of the radioactive waste and to reduce volumes.
Macro-Gap	SF11-03-15
Issue/Gap Description	Dismantling and segmentation of equipment and/or structures should be considered to reduce radioactive waste volumes and to yield an improved packaging efficiency. This requirement is not explicitly identified in the Bruce Power procedures.
Rationale	Although this is not explicitly stated in Bruce Power procedures, BP-PROC-00714 Low Level Radioactive Waste Minimization is specifically designed to provide guidance on the minimization of waste volumes. This gap will be resolved as part of the update of current governance and hence unnecessary to implement as part of IIP. DCR 28585711 has been initiated to explicitly address this requirement in BP-PROC-00714. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF11_CSA N292.3-14_9.2.6_16
Document ID	CSA N292.3-14
Article/Clause	9.2.6 Dismantling and segmentation
Requirement Assessed	Dismantling and segmentation of equipment and/or structures should be considered to reduce radioactive waste volumes and to yield an improved packaging efficiency. Note: A variety of techniques can be used depending on factors such as the material of construction, the size and shape of the equipment, and the degree of contamination. Tools used include hand tools, saws, shears, impact tools, and cutting tools. Highly contaminated portions of the equipment and/or structures may be removed to facilitate better management of the radioactive waste and to reduce volumes.
Macro-Gap	SF11-02-16
Issue/Gap Description	Dismantling and segmentation of equipment and/or structures should be considered to reduce radioactive waste volumes and to yield an improved packaging efficiency. This requirement is not explicitly identified in the Bruce Power procedures.
Rationale	Although this is not explicitly stated in Bruce Power procedures, BP-PROC-00714 Low Level Radioactive Waste Minimization is specifically designed to provide guidance on the minimization of waste volumes. This gap will be resolved as part of the update of current governance and hence unnecessary to implement as part of IIP. DCR 28585711 has been initiated to explicitly address this requirement in BP-PROC-00714. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF11_SF11 RT_5.4_15
Document ID	SF11 RT
Article/Clause	5.4
Requirement Assessed	Review task 3 examines maintenance, testing and inspection procedures.
Macro-Gap	SF11-05-15
Issue/Gap Description	BP-PROC-00498 on Condition Assessments is out of date and has been committed for future revision. The procedure needs to be updated or superseded by existing procedures which adequately capture the necessary information.
Rationale	Bruce Power reaffirmed that this procedure is to be incorporated into the aging management suite of procedures under the Equipment Reliability Program, BP-PROG-11.01 and the items identified during the inspection will be considered during these revisions per references: NK21-CORR-00531-11913 / NK29-CORR-00531-12294, Action Item 2014-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2104-002, K. Lafrenière to F. Saunders, CNSC Letter, February 4, 2015 NK29-CORR-00531-12570 / NK21-CORR-00531-12206, Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002, Bruce Power Letter, F. Saunders to K. Lafrenière, September 16, 2015 This gap is categorized as “Closed” in the database, referencing the associated ARs as well as the related assignments including Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002 and its associated ARs (Action Requests).

Gap #	SF11_SF11 RT_5.4_16
Document ID	SF11 RT
Article/Clause	5.4
Requirement Assessed	Review task 3 examines maintenance, testing and inspection procedures.
Macro-Gap	SF11-04-16
Issue/Gap Description	BP PROC-00498, Condition Assessment of Generating Units in Support of Life Extension, was last updated in 2011 and is in need of revision as part of the standard procedure document review cycle.
Rationale	Bruce Power reaffirmed that this procedure is to be incorporated into the aging management suite of procedures under the Equipment Reliability Program, BP-PROG-11.01 and the items identified during the inspection will be considered during these revisions per references: NK21-CORR-00531-11913 / NK29-CORR-00531-12294, Action Item 2014-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2104-002, K. Lafrenière to F. Saunders, CNSC Letter, February 4, 2015 NK29-CORR-00531-12570 / NK21-CORR-00531-12206, Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002, Bruce Power Letter, F. Saunders to K. Lafrenière, September 16, 2015 This gap is categorized as “Closed” in the database, referencing the associated ARs as well as the related assignments including Action Item 2014-07-4687: CNSC Type II Inspection – Condition Assessment Inspection – BRPD-AB-2014-002 and its associated ARs (Action Requests).

Gap #	SF12_CSA N290.12_5.2.2_16
Document ID	CSA N290.12
Article/Clause	5.2.2
Requirement Assessed	Procedure development personnel may participate in HF in design evaluation activities.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that procedure development personnel may participate in HF in design evaluation activities.
Rationale	This is not considered a requirement in the standard per the definition of the usage of 'may' which states may is "used to express an option or that which is permissible within the limitation of the standard". It is judged that the inclusion of a requirement for procedure writing personnel input into a Human Factors review is not warranted and can be done on an as required basis. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_CSA N290.12_5.3.2_16
Document ID	CSA N290.12
Article/Clause	5.3.2
Requirement Assessed	Training development personnel may participate in HF in design evaluation activities.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that training development personnel may participate in HF in design evaluation activities.
Rationale	This is not considered a requirement in the standard per the definition of the usage of 'may' which states may is "used to express an option or that which is permissible within the limitation of the standard". It is judged that the inclusion of a requirement for procedure writing personnel input into a Human Factors review is not warranted and can be done on an as required basis. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_CSA N290.12_5.5_16
Document ID	CSA N290.12
Article/Clause	5.5 Interface with staffing
Requirement Assessed	HF in design should consider staffing information to ensure that human actions are completed safely and efficiently for the full range of plant conditions and tasks, including a) characteristics of people who carry out tasks associated with the system being designed; --- Note: Characteristics include the qualifications, experience, skills, knowledge, training, anthropometry, gender, fitness, strength, and age of each type of system user or stakeholder. --- b) minimum staff complement; c) staffing levels and staffing goals; and d) impacts of shift schedules.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation to suggest that HF in design considers impacts of shift schedules.
Rationale	Shift schedules are a possible consideration during Background data gathering as noted in appendix B of DPT-PDE-00013 R009. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_NUREG-0700_Part_II_10_15
Document ID	NUREG-0700
Article/Clause	Part_II_10
Requirement Assessed	Guidelines for reviewing Communication Systems
Macro-Gap	SF12-02-15
Issue/Gap Description	It is not clear whether the existing communication system aligns with the intent of applicable NUREG-0700 clauses or other modern standards or guidelines.
Rationale	The current system design has proven to be fit for purpose and effective. In addition, Bruce Power current procedures as applicable to design of communication systems have been updated align with modern codes and standards. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_NUREG-0700_Part_II_10_16
Document ID	NUREG-0700
Article/Clause	Part_II_10
Requirement Assessed	Guidelines for reviewing Communication Systems
Macro-Gap	SF12-04-16
Issue/Gap Description	It is not clear whether the existing communication system aligns with the intent of applicable NUREG-0700 clauses or other modern standards or guidelines.
Rationale	The current system design has proven to be fit for purpose and effective. In addition, Bruce Power current procedures as applicable to design of communication systems have been updated align with modern codes and standards. This gap is categorized as "Closed" in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_NUREG-0700_Part_II_4_15
Document ID	NUREG-0700
Article/Clause	Part_II_4
Requirement Assessed	Guidelines for reviewing alarm system.
Macro-Gap	SF12-02-15
Issue/Gap Description	It is not clear whether the existing alarm system aligns with the intent of NUREG-0700 or other modern standards or guidelines.
Rationale	DPT-PDE-00013 (R09) is updated to account for design guidance including annunciation systems. Appendix J shows a list of inputs into Human System interfaces and design guides are part of this. Appendix T (2) discusses the general hierarchy of these inputs. Appendix Q specifically notes NUREG-0700 as "additional" design guidance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_NUREG-0700_Part_II_4_16
Document ID	NUREG-0700
Article/Clause	Part_II_4
Requirement Assessed	Guidelines for reviewing alarm system.
Macro-Gap	SF12-04-16
Issue/Gap Description	It is not clear whether the existing alarm system aligns with the intent of NUREG-0700 or other modern standards or guidelines.
Rationale	DPT-PDE-00013 (R09) is updated to account for design guidance including annunciation systems. Appendix J shows a list of inputs into Human System interfaces and design guides are part of this. Appendix T (2) discusses the general hierarchy of these inputs. Appendix Q specifically notes NUREG-0700 as "additional" design guidance. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_NUREG-0700_Part_II_5_15
Document ID	NUREG-0700
Article/Clause	Part_II_5
Requirement Assessed	Guidelines for reviewing Safety Function and Parameter Monitoring System
Macro-Gap	SF12-02-15
Issue/Gap Description	Final upgraded SSMC system did not incorporate recommendations that may have contributed to alignment with NUREG-0700 guidelines.
Rationale	The current system followed Rev08 of DPT-PDE-00013. Future design changes will follow the newer revisions of DPT-PDE-00013 in effect as required which aligns with modern codes and standards. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_NUREG-0700_Part_II_5_16
Document ID	NUREG-0700
Article/Clause	Part_II_5
Requirement Assessed	Guidelines for reviewing Safety Function and Parameter Monitoring System
Macro-Gap	SF12-04-16
Issue/Gap Description	Final upgraded SSMC system did not incorporate recommendations that may have contributed to alignment with NUREG-0700 guidelines.
Rationale	The current system followed Rev08 of DPT-PDE-00013. Future design changes will follow the newer revisions of DPT-PDE-00013 in effect as required which aligns with modern codes and standards. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF12_SF12 RT_5.1_16
Document ID	SF12 RT
Article/Clause	5.1
Requirement Assessed	The review of human factors (HF) will consider the procedures and processes in place at the nuclear power plant to ensure: a) Adequate staffing levels exist for operating the plant, with due recognition given to absences, shift working and restrictions on overtime
Macro-Gap	SF12-01-16
Issue/Gap Description	A review of internal self assessments on hours of work suggests that Bruce Power is maintaining staffing levels but not without violations that do not seem to be decreasing overall. Therefore, while programs for ensuring adequate staff levels are adequate, they are not being effectively implemented.
Rationale	This is a process implementation effectiveness issue which can be implemented through Bruce Power's current governance. This gap is categorized as “Closed” in the database, referencing the associated ARs as well as the related assignments.

Gap #	SF13_CNSC REGDOC 2.3.2_4.3_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	4.3 Other considerations
Requirement Assessed	Additional important elements that should be considered in the development of an IAMP include equipment and instrumentation, organizational responsibilities, and communication interfaces.
Macro-Gap	SF13-03-15
Issue/Gap Description	Additional important elements that should be considered in the development of an IAMP include equipment and instrumentation, organizational responsibilities, and communication interfaces. .... From 4.3.1 BDBAs and severe accidents potentially create harsh environments with high temperature, high pressure, high radiation level, and high concentration of combustible gases. These environmental conditions, which could well exceed those of DBAs used for equipment qualification, present additional challenges to the equipment. The licensee should perform equipment survivability assessments to provide reasonable assurance that equipment used in SAM is available at the time it is called upon to perform............ The habitability of the facilities used in accident management (such as the main control room, the secondary control room, and the emergency response facilities, including a technical support centre) should be assessed and assured, taking into account the environmental conditions (e.g., radiological conditions and other conditions related to lighting, ventilation, temperature and communication) within and surrounding the facilities during an accident. Where necessary, alternate habitable facilities should be provided.
Rationale	This gap is already covered under letter from F. Saunders to K. Lafreniere, Integrated Implementation Plan for Bruce A, Bruce B and Center of Site in the Next Licence Period, NK21-CORR-00531-11567, NK29-CORR-00531-11950, NK37-CORR-00531-02288, dated October 31, 2014. See GIO-011 titled “Implement enhancements to SAMG”. SIP-11: Fukushima Response - Severe Accident Management Enhancements The purpose of this initiative is to enhance the existing understanding of severe accident phenomena and SAMG capabilities. This project has a generic component, undertaken under COG Joint Project 4426 followed by station-specific implementation at each station. The scope of the work involves the following: • Enhancement of SAMG to include multi unit events and IFB events.- FAIs 3.1.2 and 3.1.3 were closed

	• Assessment of instrument and equipment survivability under severe accident and identification of equipment upgrades required- FAI 1.8.1.1 was closed • Assessment of plant habitability under severe accident conditions and identification of modifications required.- FAI 1.9.1 was closed. Improvement to understanding of severe accident phenomena including containment integrity, hydrogen production, aerosol behaviour, and in vessel retention. References: Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 9 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12828/NK29-CORR-00531-13279/ NK37-CORR-00531-02560 dated June 26, 2016 This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF13_SF13 RT 2016_5.1_16
Document ID	SF13 RT 2016
Article/Clause	5.1
Requirement Assessed	An overall review will be performed to check that emergency planning at the plant continues to be satisfactory and to check that emergency plans (EPs) are maintained in accordance with current safety analyses, accident mitigation studies and good practices.
Macro-Gap	SF13-01-16
Issue/Gap Description	Basis for minimum shift complement and ability to respond to multi-unit events- CNSC Type II Inspections of the Fall 2013 emergency exercise [70] also identified various issues for Bruce Power follow-up relating to the validation process for Emergency Mitigating Equipment (EME) guidance, and execution of key operator actions during emergency exercises
Rationale	This gap is already covered under letter from K. Lafreniere to F. Saunders Bruce Power - Minimum Shift Complement Licensing Basis and Validation, Closure of Action Item 080702 e-Doc 5022183. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_SF15 RT_5.1.3.3_15
Document ID	SF15 RT
Article/Clause	5.1.3.3
Requirement Assessed	1. Reactor Design Features for Radiation Protection 1.3.3 Radiation Protection Program Review
Macro-Gap	SF15-05-15
Issue/Gap Description	There are instances of unclear standards in the RP Program, and current RP practices are not always documented in RP Program governance. Some deficiencies were identified during investigation of the airborne alpha contamination that occurred late in 2009, and the resulting internal exposures. These deficiencies were related to the guidance provided in Clause VI.C1 (Radioactive Contamination Control). Many of the procedures related to contamination control were revised in the aftermath of that incident through the Alpha Recovery Program. However, BP-RPP-00023, Hazards Surveys, Posting, Response and Recording [89], has not been revised to include actions that should be taken upon first discovery of airborne radioactivity to contain it. This recommendation was captured in DCR 28416907, which is at "Approved" status, with a due date of March 31, 2015. However, the current revision of the procedure (R011) was issued September 25, 2014, without the required changes.
Rationale	This gap is an enhancement to the RP Program, not safety significant and can be dealt with through the Corrective Action process. The procedure is currently under revision and an AR will be entered to ensure actions to be taken upon first discovery of airborne activity is documented in BP-RPP-00023. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_SF15 RT_5.6.1_15
Document ID	SF15 RT
Article/Clause	5.6.1
Requirement Assessed	6. Radiation Protection Program Documentation 6.1. Radiation Protection Program Review
Macro-Gap	SF15-05-15
Issue/Gap Description	There are instances of unclear standards in the RP Program, and current RP practices are not always documented in RP Program governance. RP Program documentation is not always updated to reflect improvements and current practice.
Rationale	This gap is due to less than adequate implementation of the current governance and procedures and not safety significant. Timely updates to the RP Program Documentation to reflect improvements and current practices will be implemented through the Corrective Action process. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_I.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	I.C2.
Requirement Assessed	A unique by-product of nuclear electric generating station operations is the creation of highly radioactive material. If not controlled carefully, this material can adversely affect the health of individuals exposed to hazardous levels of radiation, contaminate station areas and the environment and inhibit plant access for operations and maintenance work. The radiological protection manager holds a position of major responsibility and trust for the health and safety of nuclear workers, the public, and the environment. Foremost, this manager provides leadership by setting high standards for performance and technical excellence, while creating a safety culture with a conservative approach to radiological health and safety. The radiological protection manager is an advocate for radiation safety. This individual's values, beliefs, and advocacy for radiological safety-as demonstrated by words and action-will shape the radiological protection organisation's beliefs and performance (and, more broadly, overall station performance). The radiological protection manager advises station management on radiological risk and consequences and champions initiatives that will reduce the radiation source term and minimise collective dose. The effective implementation of radiological protection activities crosses organisational boundaries. Due to this, the radiological protection manager will instil ownership for performing radiological protection activities to high standards throughout the station organisation. These guidelines can help the manager make decisions that will have a positive, long- lasting effect on the operation of the nuclear plant and result in increased radiological health and safety. The best utilities are not satisfied with the status quo, but frequently seek out the best industry practices, set challenging goals, monitor and measure, and then provide positive reinforcement to continually improve overall performance.
Macro-Gap	SF15-05-15
Issue/Gap Description	There are instances of unclear standards in the RP Program, and current RP practices are not always documented in RP Program governance. There is a gap in the effective identification of the individual and role associated with the responsibilities of the RP Manager as identified in the WANO guideline.
Rationale	This gap is due to less than adequate implementation of the current governance and procedures and not safety significant. The Program document has been revised and is in routing for approval, as part of the implementation plan, ARs will be entered to include identification of the individual and role associated with the responsibilities of the RP Manager as identified in the WANO guideline. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the

	database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_I.C4._15
Document ID	WANO GL 2004-01-R1
Article/Clause	I.C4.
Requirement Assessed	a. Management standards Radiological protection managers set high standards and expectations that are incorporated into policies and procedures. Standards and expectations are clear, concise, and relevant. They include expectations for radiological protection personnel to make conservative decisions, take action and implement changes that contribute to worker radiological health and safety. b. Programme monitoring Goals and objectives for the radiological protection organisation should support corporate and station goals and objectives while addressing areas where performance improvements are needed. Radiological protection managers establish measurable, achievable, and challenging radiological protection goals to improve performance. Typical performance indicators, such as those noted below, are used as tools to monitor and trend performance. o Station outage and on-line collective dose; o Unplanned internal dose greater than 0.1 mSv; o Individual and work group dose; o Number of radiological hot spots; o Radiation source term, as measured on out-of-core piping (boiling water reactor radiation assessment and control [BRAC] points; PWR Electric Power Research Institute [EPRI] standard radiation monitoring programme [SRMP]); o Radiation source term, as measured from the station chemistry effectiveness indicator (CEI); o Occurrences of unplanned individual dose above administrative or radiological work permit control levels; o Control of high radiation areas (HRAs), as measured by the number of occurrences of unposted HRAs or unauthorised entries into HRAs; o Leak containment devices installed on contaminated systems; o Amount of recoverable plant area contaminated Exempted areas include the following: - Locked high radiation areas; - High radiation areas where ALARA comparison of estimated dose savings is less than estimated dose to decontaminate the area; - Areas that by nature are not intended to be decontaminated, such as decontamination rooms, sample sinks, fume hoods and downdraft tables; - Areas posted as contaminated for outage activities several weeks prior to the outage through several weeks following the outage. o Skin and clothing personnel contamination events by EPRI action levels 1, 2, and 3; o Positive whole-body counts;

	o Instances of uncontrolled radioactive material found either outside the radiologically controlled areas (RCAs) or outside the protected area; o Instances in which contaminated individuals or material were detected at the protected area exit portal monitors; o Electronic dosimeter accumulated dose alarms; o Unanticipated valid electronic dosimeter dose rate alarms; o Human-performance-related improper radiological work practices; o Ratio of self-identified problems versus problems identified by others; o Performance of portable and fixed radiological survey instruments, to include metrics such as calibration and source-check failures; o Performance of the primary dosimeter system, as measured by quality control testing of primary dosimeter reading bias outside the control band; o Radioactive waste volume generation; o Gaseous and liquid effluent activity; o Number of high radiation and locked high radiation areas; o Number of outside radioactive material storage areas that are unprotected from the elements; o Number of personnel dose extensions beyond administrative dose control level. Other performance measures can be developed and used to improve performance in a specified area.
Macro-Gap	SF15-05-15
Issue/Gap Description	There are instances of unclear standards in the RP Program, and current RP practices are not always documented in RP Program governance. RP Program documentation is not always updated to reflect improvements and current practice.
Rationale	This gap is due to less than adequate implementation of the current governance and procedures and not safety significant. Timely updates to the RP Program documentation to reflect improvements and current practices will be implemented through the Corrective Action process. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_I.C5._15
Document ID	WANO GL 2004-01-R1
Article/Clause	I.C5.
Requirement Assessed	The radiological protection manager strongly instils and periodically reemphasises principles of radiological health and safety. Pressures to reduce cost must not affect the conservative decisions needed to ensure radiological health and safety. Work schedules must not compromise radiation protection standards and controls. Personnel must not feel a sense of pressure to proceed in the face of uncertainty, or to compromise radiological protections standards, to meet schedules. Clearly establish management expectations and guidance for reacting in a conservative manner when faced with uncertain or unexpected radiological conditions. Communicate management's support for conservative decision-making by personal example and in clear, unequivocal terms. Frequently reinforce this through training, observing field activities and coaching. Use industry and plant operating experience to demonstrate vulnerability to similar events. Stress the importance of recognising activities that increase the risk of a radiological event. Reinforce conservative decision-making principles, including the following, to radiological protection personnel at all levels: o Question and validate available information; o Do not proceed in the face of uncertainty; o Involve supervision; o Recognise when degraded conditions exist that could affect radiological health and safety; o Gather and analyse information from relevant sources and key stakeholders to clearly define and provide options for problem resolution; o Use all available technical resources, including people off site if necessary; o Critically and objectively consider the short- and long-term radiological risks, consequences and aggregate impact associated with the various decision options; o Develop implementation plans that include contingencies and compensatory measures to maintain and enhance radiological health and safety; o Clearly identify decision-makers and their roles and responsibilities; o Communicate the bases for the decisions throughout the organisation. Refer to the following WANO documents for additional information: o WANO GL 2002-01 Principles for Effective Operational Decision-Making o WANO GP ATL 08-003 Human Performance Tools for Managers and Supervisors
Macro-Gap	SF15-01-15
Issue/Gap Description	The Bruce A and Bruce B ALARA Committee and Sub-committee TOR does not include reference to adhere to BP RPP-00044, the required

	meeting agenda items, or timelines for minute distribution
Rationale	This gap is due to less than adequate implementation of the current governance and procedures and not safety significant. An AR will be entered to update Bruce A and Bruce B ALARA Committee and Sub-committee TOR to include reference for adherence to BP-RPP-00044, the required meeting agenda items, or timelines for minute distribution This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_IV.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	IV.C2.
Requirement Assessed	a. Airborne radioactivity surveys 1) Sample frequency and collection methods Airborne radioactivity surveys are performed to monitor the concentrations of airborne radioactivity associated with nuclear station operation. They are to be performed as follows: o During any work or operation known or suspected to cause airborne radioactivity, such as grinding, welding, burning, cutting, hydrolyzing, vacuuming, sweeping and using compressed air or volatiles on contaminated equipment; during waste-compacting operations; and during contaminated insulation removal; o During any work or operation that involves the breach of a radioactive system for which the potential for measurable airborne radioactivity exists; o Prior to or during initial entry into any known or suspected airborne radioactivity area or area with significant loose surface contamination (for example, >=100,000 dpm/100 cm2), and periodically thereafter; o Containment/drywell entries if conditions are unknown; o Prior to or during initial entry into any high-risk area such as steam generators, reactor cavities, reactor vessels, or radioactive waste tanks, and periodically thereafter; o Based on environmental factors, such as dry and dusty conditions or the drying out of highly contaminated areas, components, and filters; o When the potential for airborne activity exists, such as the discovery of a significant spill or spread of radioactive materials; o Periodically (such as daily) in RCAs with the potential for changes in airborne radioactivity, including the containment or drywell during outages o Any time respiratory protection devices or alternate tracking methods (DAC-hours) are used to control internal radiation dose; o During any work or operation over or near the spent fuel pool when the coolant analysis indicates elevated levels of tritium; o More frequently when analysis of the reactor coolant indicates the presence of significant fuel leaks, which raises the potential to encounter alpha activity. Increases in gamma-emitting fission products such as the cerium, ruthenium, barium, lanthanum and americium as well as noble gases, can indicate that alpha emitters have been introduced into the coolant. Also, evaluate previous fuel failures and alpha contamination history, because alpha contamination may be trapped in crevices or surface corrosion layers. Obtain a representative sample of the air breathed by personnel in the area. Use low-volume air sampling to determine airborne radioactivity levels for worker protection. Use high-volume air sampling for situations in which airborne radioactivity concentrations need to be determined rapidly;

	when the work being monitored is not of sufficient duration to support the time requirement for low-volume air samples; or in conjunction with low-volume air samplers to determine peak airborne concentrations. Lapel air samplers can also be used to obtain representative samples of the worker's breathing zone and are required for work in alpha level 3 areas. When selecting the air sampler location, consider the effect of air flow from plant or temporary ventilation on the sampler's ability to collect a breathing zone air sample. Take air samples during the expected periods of highest concentration, and evaluate them as quickly as possible to determine the need for engineering controls, respirators, area evacuation, area posting, and worker relief from unnecessary respirator use. Use continuous air monitors (CAMs) for situations in which airborne radioactivity levels can fluctuate, and early detection of airborne radioactivity could prevent or minimise radioactivity inhalation. The monitors should also be located near plant systems that could cause rapid increases in airborne radioactivity, such as the recombiner, offgas, steam jet air ejector, or other steam-related systems in a BWR and the refuelling floor during refuelling evolutions. Use CAMs with iodine detection capability when removing the reactor head and internals as well as during the initial opening of BWR steam systems (such as main steam reheaters). Periodically sample and analyse plant liquid systems that could concentrate tritium. Conduct bioassays when significant tritium intake could occur (for example, following entry into a tritiated steam atmosphere). 2) Equipment setup and calibration Equipment used for the sampling and monitoring of airborne radioactivity is maintained in good working order, and is periodically checked to verify accuracy. For example, check the proper operation of CAMs periodically by checking for instrument response to a radiation source. Also, monitor the airflow and airborne activity readings periodically while personnel are working in the area. Air sampling equipment with inlet extension hoses, including continuous air monitors, should not be used for quantitative evaluation of airborne radioactivity levels unless the length, diameter, material, layout, and condition of such hoses has been analysed to show that excessive particle deposition will not occur in the extension line. If inlet extension lines are used for quantitative assessments, adjust the alarm setting of the continuous air monitor to account for line deposition. Set CAM alarm levels to alarm consistently at two or three times the background count rate. CAM alarm set points may be raised during periods of high short-lived fission product or radon progeny product concentrations, with the approval of radiological protection supervision. Document set point changes, so that they can be returned to normal when short- lived or natural radioactivity is no longer significant. Check alarm capabilities and set points periodically (typically done daily) to ensure proper operation. Calibrate air sampling equipment annually at the very least. During operation, closely monitor air sampler flows and activity readings. Air samplers that exhibit low or rapidly oscillating flow, erratic or off-scale

	activity readings, indications of air flow leakage around filters or other indications of damage, should be removed from service, repaired and recalibrated. Test counting equipment daily for accuracy and use charts to trend system response. Perform efficiency calibrations with isotopes which correspond to the station radionuclide mix. 3) Sample analysis and review Analyse airborne radioactivity samples as follows: o To rapidly screen air samples, measure each sample with a thin-window Geiger-Mueller (G-M) detector. Alternatively, a more detailed measurement of activity, especially low-level activity, may be made using a G-M detector and a scaler, or a gas flow proportional counter, both of which will detect beta radiation. Screening methods should consider isotopic mix, sample geometry, and count time. If airborne activity is detected above 0.30 DAC, take protective actions to minimise personnel dose while a radionuclide analysis is performed. O A radionuclide analysis of the sample is achieved with a high-resolution gamma spectrometer (for example, germanium). Air samples suspected to be greater than 0.30 DAC should be counted using such a system which assesses the types and quantities of radionuclides accurately. O Alpha, transuranic and other hard-to-detect radionuclides, are often significant contributors to dose from airborne radioactivity. Develop and use methods to account for these radionuclides in the assessment of airborne radioactivity. Ratios can be developed based on representative reactor coolant sample activity and waste stream analysis data. Evaluate changes in plant operation that could significantly alter the isotopic mix. For example consider fuel failures since the last waste stream analysis was performed, and the need to resample and reanalyse for alpha and hard-to-detect radionuclides. It is important to use chemistry sample results to anticipate radiological conditions that may impact radiological controls. Record the results of airborne radioactivity surveys. Include details about: o The date and time the air sample was taken; o The purpose and location of the sample; o The applicable RWP; o The amount of air sampled; o The results of sample counting; o The serial number of the air sampler and counter used; o The name of the person who obtained and counted the sample. Radiological protection supervision should review air monitoring surveys in a timely manner to verify calculations and identify trends in airborne radioactivity levels. A. Posting and access control Use postings and controls to minimise exposure of personnel to airborne radioactivity. Areas of airborne radioactivity concentration greater than 0.30 DAC, should be conspicuously posted. Require a radiation work permit, work procedure or access permit for entry

	into an airborne radioactivity area. The posted area should be enclosed or contained within a room, tent, bag, box or other device, to prevent the spread of airborne and loose surface radioactive contamination. B. Work control methods 1) Planned internal dose Establish policies and procedures for planned internal dose. These policies and procedures should utilise engineering controls to reduce airborne radioactivity and minimise internal deposition of radioactive material. A thorough evaluation of control methods, avoided dose and the overall dose to the worker is required prior to approval of planned internal dose. Workers should be informed of their planned internal dose and avoided external dose as well as the required documentation and approval levels. Have procedures in place which highlight the administrative controls required, for workers with internally deposited radioactivity, to process in and out of the RCA; the frequency of whole-body counts; the methods used to differentiate between external contamination and internal activity; and inhalation versus ingestion. 2) Engineering controls Engineering controls are preferred over the use of respirators to minimise internal dose. Respirators can cause additional stress to workers and increase the risk of injury by interfering with vision, freedom of motion and the ability to communicate. These factors may also contribute to increased dose from external sources. Therefore, engineering controls should be fundamental to work planning and be used as much as possible to minimise internal dose. Only when further engineering controls are impractical and the use of respirators is shown to minimise total dose, should respirators be considered. Include potential negative post job impacts, such as the need to collect alpha bioassays, or the impact of personnel with internally deposited radionuclides alarming contamination monitors, in the decision on respirator use. The radiological protection group periodically assesses engineering controls being used to control airborne radioactivity. This assessment should include the following: o The use of portable or fixed ventilation devices to reduce or eliminate airborne radioactivity concentrations; o Decontamination and/or repair of the source of airborne radioactivity; o Containment of the source, or the potential source, of airborne radioactivity (for example, use of contamination containments or glove bags); o Performance of the work under water, exposed surfaces being kept wet and the use of fixative agents; o Installation of permanent engineering controls in areas where airborne radioactivity is expected; o Comparison of dose saved when engineering controls are installed. 3) Respiratory protection When engineering controls cannot be used to reduce airborne radioactivity

	to appropriate levels, the assessment also includes the following: o The total dose with and without respiratory protection; o Past experience on similar tasks, current airborne radio- activity levels and contamination levels; o Radionuclide concentration in fluid systems; o Expected DAC-hours for the job and the number of previous DAC-hours assigned to the worker. Radiological protection supervisors review and document the results of this assessment. Consider the potential negative consequences of intakes. These could include: o Additional administrative controls required for workers with internally deposited radioactivity, to process in and out of the RCA; o The frequency of whole-body counts; o The increased challenge to radiological protection personnel to differentiate between internal activity and external contamination; o The loss of worker productivity; o The potential newsworthy nature of the event. Issue respirators only to personnel who are trained, fit-tested for the type of respirator worn and medically qualified to wear them. Maintain positive controls for the issue, use and return of respirators, to ensure only qualified personnel wear them. When plant services, or instrument compressed air systems, are used to supply air for respirators, test the air to verify that it meets regulatory requirements as well as to determine that it is free of radioactivity. Fit, check, test, clean, repair and procure respirators in accordance with regulatory requirements and recognised national standards.
Macro-Gap	SF15-02-15
Issue/Gap Description	BP-RPP-00023 Hazards Surveys, Posting, Response and Recording, has not been revised to include actions that should be taken upon first discovery of airborne radioactivity to contain it.
Rationale	THis gap is an enhancement to the RP Program, not safety significant and can be dealt with through the Corrective Action process. The procedure is currently under revision and this gap has been included as part of the updates to BP-RPP-00023. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_V.C1._15
Document ID	WANO GL 2004-01-R1
Article/Clause	V.C1.
Requirement Assessed	Plan work, including engineering design work, involving radiation dose as far in advance as practical. Optimise the ALARA principles of time, distance, and shielding for each work activity. During the planning stage, avoid unnecessary work, sequence work to minimise dose and identify the lowest dose options for performing the work. System engineers, maintenance planners, outage schedulers, and job supervisors should actively participate in all phases of dose reduction planning to ensure success. Specific steps that have proven useful include the following: o Decontaminate plant components and work areas and evaluate the need for temporary shielding prior to initiating maintenance work in the affected area. Consider chemical and mechanical decontamination techniques. o Determine needed tools and parts before the work begins, and stage them so delays are minimised. Use power tools (electric or pneumatic) to reduce task performance times wherever possible. Consider the use of special tools (including robotics and remote handling equipment). o Provide support services, including electrical, water, air, and auxiliary lighting and a working environment with comfortable temperature, humidity, and space. o Coordinate the efforts of different groups, such as Operations, Construction, Maintenance, and Radiological Protection, so work can proceed in a systematic, efficient manner. Evaluate the amount of work scheduled in an area to minimise work and work crew interferences. o Minimise the number of workers assigned to a particular job. During outages, to the extent possible, minimise the number of personnel allowed inside containment (PWRs) or the drywell (BWRs), to reduce congestion and improve work efficiency. o Coordinate work by plant area so that work such as scaffolding, insulation, shielding installation and removal is not duplicated for multiple tasks in the same area. Create an integration of scheduled activities to improve coordination. o Schedule system or component flushes to eliminate hot spots and/or reduce general area dose rates prior to work. o Review historical data for previously performed jobs and, where feasible, benchmark industry best performance both in terms of person-hours required, techniques used and dose received. o Review design changes to determine their dose impact from installation, operation and maintenance. o Evaluate engineering controls to reduce airborne activity to minimise internal dose and improve worker efficiency. o Evaluate the use of remote monitoring equipment, including teledosimetry, video monitoring, and two-way audio communication.

	o For outages, schedule enough time to clean up reactor coolant system activity so that activity will not plate out on piping surfaces and increase work area dose rates. o For outage, develop a water management plan to ensure high activity water is not transferred to systems that would result in additional dose to workers. o For outages, schedule primary system valve maintenance to follow reactor system cleanup. o Evaluate replacing components rather than repairing them at the job site, based on a cost/benefit analysis. o Schedule or sequence work such that it is performed when systems are full (not drained), to take advantage of lower dose rates (such as for steam generators, tanks, and piping). o For outages, evaluate the need for shielding or other radiological engineering activities based on the total work scope in a certain area. Although the analyses of the individual tasks may not warrant such activities, the consideration of the total scope may change the assessment. o During the planning phase, use just-in-time training, dry runs and training on realistic mock-ups under simulated field conditions to improve work efficiency. Estimate dose based on an accurate prediction of time in the radiological work area, body position relative to the source of radiation and dose rates in that area. Previous dose records for the jobs being performed, either from the station or from other stations, may be useful once adjusted for the scope of the current work and changes in dose rates. Establish a job radiation dose history file to capture this information. If previous dose records are not available, estimates can be based on time and dose rates. However, review these estimates carefully to ensure that both accurate person-hours in the RCA and dose rate values have been used. Establish a collective dose action level (in person-Sv) for when a more thorough review is conducted. Typically, this action level is 1 person-rem or less. Some stations have implemented this level of rigor as low as 0.001 person-Sv. Senior managers review tasks expected to exceed 0.05 person-Sv and tasks for which large individual doses could be received in a short time. This review should ensure that sufficient planning and resources have been applied to dose reduction. A station ALARA or dose oversight committee can be used to coordinate resources and establish priorities to achieve outage and on-line dose reduction. These committees, typically composed of department managers and chaired by station senior management, meet on a regular basis. In addition, some utilities have developed fleet ALARA committees chaired by senior department management to provide oversight of fleet ALARA initiatives and dose reduction strategies.
Macro-Gap	SF15-01-15
Issue/Gap Description	There is misalignment regarding planning dates between BP-RPP-00011, Requirements for Planning Radiological Work and BP-PROC-00342 Sheet 001, Planned Outage – Preparation Milestones.

Rationale	This gap is due to less than adequate implementation of the current governance and procedures and not safety significant. An AR will be entered to update BP-RPP-00011 to ensure that it is aligned with BP-PROC-00342 Sheet 001. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_V.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	V.C2.
Requirement Assessed	Work supervisors, in coordination with radiological protection personnel, conduct pre-job briefings to inform workers of the specific actions that are planned to reduce dose during the task. Briefings are also an opportunity for workers to input ideas for reducing dose. Review tasks estimated to exceed the collective dose action level for application of task-specific dose reduction techniques, as described below. o Track task progress daily against both the estimated person-hours or per cent complete and the estimated person-Sv to identify tasks that are at risk of exceeding projections. o Conduct in-progress reviews at pre-established intervals for tasks that exceed the station person-Sv threshold levels. o Set these designated intervals prior to the task and base them on the type of work to be performed, task duration and expected radiological conditions. o The interval can be designated as a per cent of estimated person-Sv, per cent of person-hours, or per cent complete, such as 25, 50, 75 and 100 per cent. o The interval could be established based on other logical decision points in the task. For example, in-progress reviews could be performed after initial shielding installation or after component removal but prior to installation of the new component. If in-progress reviews indicate that the dose estimate may be exceeded, consider whether additional dose reduction techniques can be used. o Also reconsider any techniques that were previously rejected as not resource efficient. o Dose reduction techniques that were used should be reviewed for effectiveness and the cause(s) for the projected overruns determined. o Following in-progress reviews, evaluate the need to adjust the dose estimate up or down to ensure that it remains accurate and challenging. o Communicate dose estimate changes and their bases to affected management, work groups and radiological protection task coverage personnel. o Include these in the outage ALARA report as lessons learned for future outages. o Perform a post task review for all work activities with specific ALARA plans. o As a minimum, for tasks with actual collective dose above an action level or actual doses exceeding an estimate (for example, by 25 %), conduct a formal cause analysis and determine corrective actions. o Most supplemental personnel leave the site shortly after a task is completed or prior to the end of an outage. Post task reviews are held as soon as practical after task completion. o Ask the workers for suggestions on how to reduce dose on future

	jobs. o A record of similar work should be kept for future reference. o Repetitive tasks performed on line can be reviewed collectively on a less frequent basis (such as quarterly). ALARA or dose reduction suggestion programmes that recognise and reward workers for practical suggestions to reduce dose have proven effective in generating new ideas. Including dose goals as part of existing incentive programmes has helped some stations achieve additional focus on dose reduction.
Macro-Gap	SF15-01-15
Issue/Gap Description	There is no documented requirement to include dose goals as part of an incentive program at Bruce Power.
Rationale	This gap is an enhancement to the RP Program, not safety significant and can be dealt with through the Corrective Action process. An AR will be entered to consider including dose goals as part of an incentive program at Bruce Power. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_VI.C3._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C3.
Requirement Assessed	a. Area contamination control 1) Source minimisation o Control the sources of radioactive contamination to minimise the number and extent of contaminated areas. o Identify radioactive system leaks and enter them into the station work control system for repair, with priority given to leaks that spread contamination. o Use drip pans, containment devices, or drain hoses to divert or collect leakage whenever the leak cannot be repaired quickly. o Track and inspect these devices periodically (for example, monthly) to ensure effective protection against the spread of contamination and timely removal after repair of the leaking components. Prepare work sites to minimise the spread of contamination during work while also reducing the generation of radwaste. Planning includes contamination control measures such as the use of plastic, washable sheets or absorbent material and the use of strippable coatings, containments or bottles to collect radioactive material leakage. Train workers to ensure they are proficient in using contamination control devices. Maintain the integrity of floor coatings to facilitate decontamination of areas after work is completed. Avoid the use of materials that attract radioactive particles or that have been known to accumulate contamination, such as cloth chairs and carpet in the RCA. If these materials must be used, survey them frequently to ensure contamination levels are not building up above appropriate levels. Provide specific contamination control guidance for repetitive evolutions such as filling and venting contaminated systems and installing temporary instrumentation during in-service testing. Document guidance in plant procedures, job aids and radiation work permits to minimise the chance of spills during these repetitive evolutions. For frequently performed fill-and-vent evolutions, consider modifications to eliminate the use of temporary hoses routed to floor drains or to ducts that create a high potential for the spread of contamination. Store material with loose or fixed contamination in areas protected from inclement weather, water leaks, extreme temperatures, fire hazards, and other environmental conditions that could degrade the material or storage container and spread contamination. Investigate spills caused by improper maintenance, surveillance testing, and operational evolutions. Use the station corrective action programme to document spills. Take appropriate corrective actions, such as establishing procedure revisions, process changes, plant modifications, and task-specific training as well as correcting improper work practices, to prevent recurrence of spills. Clean work areas and survey them for contamination after each job.

	Contain and clean spills or leaks that spread radioactive contamination as soon as practical. After fixing or containing leaks, decontaminate component rooms and cubicles to allow personnel to access them without wearing protective clothing. The use of containment enclosures (for example, glove bags or tented enclosures) can reduce post maintenance decontamination efforts, in addition to eliminating airborne contamination. Evaluate the dose received and resources needed to install and remove containment enclosures against other contamination controls. 2) Survey frequency Survey for beta/gamma radioactive contamination at a frequency appropriate for the conditions and activities conducted in a given area. The frequency and the extent of the survey are based on historical data, the potential for change and the need for reducing dose to radiological protection technicians. Surveys need not be performed in areas that are accessed infrequently. Conversely, personnel should not be allowed to enter until these areas are surveyed. Examples of survey frequency are as follows: o weekly in contaminated areas accessed frequently or in areas where radioactive materials are handled or stored o When necessary to control entry or work where contamination boundaries are located in areas of high dose rates o During initial entry into areas entered infrequently that contain known or suspected contamination areas and periodically thereafter to determine if conditions have changed o At least daily at contamination area control points, change areas, and step-off pads when in use o At least daily at RCA exit points o During work that involves the opening of any radioactive system; and during welding, burning, or grinding on surfaces with loose or fixed contamination o Following area decontamination to ensure that removable levels are less than 1,000 dpm/100cm^2 o Any time contamination conditions are subject to significant or rapid change in a work area o Routinely in areas outside the RCA (offices, shops, storage areas, and eating areas) on a rotating basis. If contamination is found in these areas, perform additional surveys to ensure that no additional contamination is outside controlled areas. Sample locations should not be restricted to general walkways for routine surveys. Obtain samples from out-of- the-way locations and equipment as well as from potential sources of contamination, to ensure a complete assessment of the area. A representative number of smears should be checked for alpha activity. Take periodic samples from primary sample sinks and in the reactor cavity following drain-down after refuelling. Conduct periodical surveys on contamination areas that have the potential for highly radioactive particles. Base survey frequency on the potential for worker contamination on contamination history, current survey results, trends and on the dose expended to perform these surveys. Areas directly

	adjacent to discrete radioactive particle areas should also be surveyed periodically during work and at a lesser frequency when work is not being performed. Standard dry-smear techniques are not sufficient to collect particles because particles frequently will not adhere to the smear. The most effective survey method is to use large-area smears taken with tape, oil-impregnated cloth, tacky rollers or similar devices. Document and retain the results of contamination surveys taken to assess the level of worker protection. Radiological protection supervisors review contamination survey results to ensure that all required surveys are performed and that documentation is accurate and complete. Trends in contamination levels that require further investigation are also identified. 3) Contamination area posting and work control Whenever practical, post contamination survey information in the form of maps, signs or stickers conspicuously in or near work areas. For work in areas with known discrete radioactive particles, consider additional precautions. Such precautions include special posting, increased contamination monitoring, the segregation of material from the area, and the use of buffer zones to prevent the spread of particles. Unless the area is bounded by walls and doors, a tent, or containment, the area should be clearly marked with an appropriate combination of yellow and magenta rope, signs, gates or boundary tape to signify the presence of radioactive contamination. Areas such as sample sinks, pump bases, and other small areas that surround equipment may require alternate methods of marking the presence of radioactive contamination. Workers must be able to determine the boundaries of the contamination area. Ensure the integrity of boundaries by prohibiting personnel from reaching across or passing material over boundaries. Secure cords and hoses that penetrate boundaries and by restricting material from encroaching on boundaries. A person with an open wound should not be allowed access into a contaminated area unless the wound has been covered to prevent contamination from entering the body through the wound. Radiological protection personnel should be informed immediately of any wound or other injury occurring in a contaminated area so the injury can be checked for contamination. Since the radiation dose from contamination is usually insignificant, actions necessary to provide prompt emergency medical attention MUST NOT be delayed by attempts to monitor for contamination. 4) Protective clothing requirements Include protective clothing requirements as well as other protective and precautionary radiological measures, in a radiological work procedure or permit. Personnel entering contaminated areas should wear protective clothing, as follows: o Protective clothing is worn based on the contamination levels and the type of work to be performed. A complete set of protective clothing normally consists of a head cover, coveralls, gloves, booties and rubber or cloth overshoes. Cotton liners can be worn underneath rubber gloves for comfort, but they should not be considered protection from contamination. If a respirator is worn, the head cover should not interfere with the seal

	between the face of the worker and the respirator. o Some plants use types of protective clothing such as scrubs instead of more traditional protective clothing coveralls in areas with lower contamination levels. Regardless of the style of clothing worn, if used as protective clothing to reduce the risk of a worker becoming contaminated, the clothing needs to be designated and controlled in the same manner as traditional protective clothing. This control includes removal at the step-off pad at the exit from the contaminated area. Scrubs used as protective clothing should not be worn outside the RCA, because of the buildup of low-level fixed contamination. Also, scrubs with low-level fixed contamination should not be stored outside the RCA in an uncontrolled area. o When scrubs are not considered protective clothing such as when used as a modesty garment under other protective clothing or when worn as street clothing in areas where personal street clothing and partial protective clothing such as gloves or shoe covers are authorised it may be acceptable for personnel to exit the contaminated area without removing the scrubs at the step-off pad. o If personnel are working in a contaminated area with significant removable contamination (for example, in excess of 100,000 dpm/100 cm2), additional protective clothing may be required. If work involving wet or greasy materials is expected or encountered, non- permeable coveralls or aprons should be used in addition to a full set of regular protective clothing, to protect personnel from wet materials. o A step-off pad is provided at the exit of a contaminated area where protective clothing is removed before personnel exit the area. This step-off pad is on the clean side of the contaminated area exit. In areas where more than one set of protective clothing is used, additional step-off pads may be used to prevent the spread of contamination. Placing receptacles on the contaminated side at step-off pads for segregating reusable protective clothing and trash reduces the potential for the spread of contamination. o The number of layers and type of protective clothing may be adjusted based on other industrial safety risks, such as heat stress. The goal should always be to optimise worker protection and to prevent the worker from becoming contaminated. A reduction in protective clothing requirements for industrial safety reasons warrants compensatory actions to further reduce or contain work area contamination. The effectiveness of protective clothing is greatly reduced when dampened from perspiration or moisture from the work environment. Precautions, such as the use of air chillers and dehumidifiers, should be used to control these factors. Alternatively, frequent changes of protective clothing may be required. 5) Radiologically controlled area posting and work control. Controls for the RCA include the following: o The area is marked conspicuously. o Personnel normally are not allowed to eat, drink, smoke and chew in the RCA. If necessary, techniques for providing water to personnel can be used if precautions including contamination control and monitoring are

	taken to minimise the potential to ingest radioactivity. o Each entry into the RCA is controlled by a radiation work or access permit. o Personnel normally are not allowed to exit a contaminated area and traverse the RCA in potentially contaminated protective clothing. o Uncontaminated areas within the RCA should be kept as clean as practical. o When a significant fraction of smears from an area indicates loose contamination above a designated administrative control level, the area is cleaned. o The extent and status of station contaminated areas are tracked, and periodic reports are sent to management. b. Discrete radioactive particles Discrete radioactive particles (referred to as DRPs or "particles" hereafter) are small, loose, highly radioactive particles that are very transportable because of their small size and electrostatic charge. Particles originating from irradiated fuel emit high-energy betas and low-yield photons, resulting in high beta dose rates. Particles originating from activated corrosion products emit low-energy betas and high- yield, high-energy gammas, resulting in high gamma dose rates. Evaluate technical and operational considerations and develop a failed fuel action plan or procedure for operating with the defective fuel. This plan or procedure should include the added potential for the production of DRPs. Minimise the generation and spread of particles during maintenance activities that involve the opening of primary systems. Proven techniques for reducing DRPs during in- place valve seat maintenance include installation of dams in valves and piping prior to maintenance and then vacuuming the inside of the valve after maintenance and wiping the inside of the valve with a wet towel. Techniques such as X-ray fluorescence can be used to determine cleanliness more accurately than visual inspection. Carefully monitor refuelling equipment used at other facilities before allowing its entry into the plant. After use, clean and carefully monitor the equipment before it is allowed to cross the plane of the pool edge. When particles are known to be in the fuel pool, use submicron underwater filters and fuel pool skimmers to reduce the concentration of particles both in the pool and attached to the pool walls at the water surface. The use of underwater vacuum cleaners has proven effective in reducing particles in spent fuel pools and flooded reactor cavities. Entry into areas with known or a high potential for DRPs requires specific radiation work permits and increased radiological protection controls, including additional protective clothing. Outer protective clothing layers should be either discarded after use or handled separately to avoid cross-contamination of other less-contaminated clothing. In addition to their normal task coverage functions, radiological protection technicians' responsibilities for work in an area with a high potential for DRPs include the following: o Establishing stay times in the work area based on the potential for significant exposure from DRPs.

	O Surveying materials and equipment for the presence of DRPs before use by workers. O Periodically surveying workers in the area based on the potential for significant exposure from DRPs. O Assisting or observing workers during removal of outer protective clothing to help avoid contamination of inner protective clothing and the workers. O Wiping down respirators to remove discrete particles prior to bagging and removal from the area. O Clearly identifying material removed from DRP areas, such as tools and radioactive waste. O Using specific colored bags for all materials removed from DRP areas, and prohibiting opening except in specially equipped areas. Protective clothing is removed at the boundary of the DRP area and a whole-body frisk is performed as close to the DRP area as possible. Contamination monitoring using a whole-body frisker is performed as soon as workers leave the DRP area. The monitoring can also be done before workers are allowed re-entry to the DRP area. Develop procedures for and train radiological protection personnel in identifying particles. These procedures should describe decontamination methods, which include the capture of the particle for later analysis and the correct survey methods to aid in subsequent dose determinations. Incorporate skin dose calculation methods (for example, VARSKIN or equivalent) specific to small particles of radioactivity into plant procedures. C. Equipment and material control Minimise long-term on-site storage of low-level waste (LLW) 1) Surveys Survey equipment and material being transferred from RCAs, contaminated areas and highly contaminated task locations for loose and fixed contamination. Ensure that limits are met and that no detectable radioactive material is unconditionally released from the protected area. Potentially contaminated bulk materials such as soil shall be analysed and determined to be free of detectable contamination prior to release. For bulk materials that are not suitable for normal loose and fixed contamination level assessment techniques, count representative samples to environmental levels using established procedures and methods. This is done to ensure that no detectable radioactive material is unconditionally released with the bulk material. For unconditional release surveys of equipment and material, exercise caution to ensure the item is surveyed by qualified radiological protection personnel. Dismantle the equipment or use special survey techniques to gain access to inaccessible surfaces for monitoring. Treat inaccessible surfaces as contaminated unless an evaluation determines that no potential exists for contamination. Consider an independent survey or supervisory approval if materials are released by survey with a handheld instrument. Some utilities have established logs to document unconditional release surveys, to ensure a sense of personal accountability and to help identify the source of any radioactive material found outside the RCA.

	Develop a release plan for those items going into the RCA that cannot be surveyed in a gamma-sensitive tool monitor and that are expected to be released from the RCA. The use of automated gamma-sensitive tool monitors eliminates human error normally associated with manual frisking and improves the ability to detect contamination composed primarily of gamma emitters. Station workers may be trained to use the automated tool monitors for personal items that have not been taken into contaminated areas. In general, the following are the recommended methods for monitoring personal items for removal from the RCA: o Lanyards, hard hats, badges and primary and secondary dosimetry may remain on the individual and be worn through the whole-body contamination monitors. O Sensitive items worn by security personnel, such as firearms and ammunition, may remain on the individual and be worn through the whole-body contamination monitors. O Personal items in an individual's pockets or worn on the belt, such as mobile phones, pagers and keys, may remain on the individual and be worn through the whole-body contamination monitor. These items should be monitored in the gamma- sensitive tool monitor if used in the RCA. O Certain items should always be released by monitoring in a gamma-sensitive tool monitor. These include the following: - data logging devices in the RCA (for example, operator rounds data loggers) - radios - flashlights - gloves - hand-carried items, such as notebooks, pens, and briefcases Trained qualified radiological protection technicians perform unconditional release surveys of equipment and tools. Personal tools that have been used in a contaminated area, which are typically worn on the belt - such as multi-tools, fuse pullers, and pocket knives - are included in such surveys. Prior to unconditional release, ensure that items are free of loose surface contamination, either by process controls and/or physical surveys. Process controls include maintaining radioactive contamination within established boundaries, routinely surveying uncontaminated areas within the RCA and being aware of the areas the item was used in prior to release from the RCA. All tools, equipment and items removed from contaminated areas must be surveyed by trained RP technicians. 2) Container controls Equipment and material with contamination limits above control levels are stored in contaminated areas or radioactive material storage areas after being placed in containers. Radioactive waste containers that will not be opened on site do not require documentation of internal contamination levels. Containers that are continuously attended by a radiation worker need not be labelled, such as at a drum packing station or while materials are being loaded into a second container. Whenever practical, use strong, tight containers. Seal bags with tape. Material with fixed contamination may not need to be

	placed in containers but may still need to be labelled and controlled. 3) Vacuum cleaners Control vacuum cleaners used within the RCA. Effective controls include the following: o Vacuum cleaners used in RCAs have high- efficiency particulate air (HEPA) filters installed to filter the exhaust. O A filter integrity test is performed following installation of a HEPA filter to ensure that the filter is in good condition and is installed properly. The test is repeated annually or when activities such as opening of the vacuum could have invalidated the test results. O Vacuum cleaners designated for RCA use are controlled such that only authorised/trained personnel can access or operate them. For example, vacuums are locked in controlled cages or rooms, or electrical plugs and air inlet connections are secured with locking devices controlled by RP. O Personnel who open vacuums are trained on proper contamination controls, filter installation and inspection. O A locking device or seal is employed on joints between the vacuum cleaner head and body to prevent inadvertent opening of the unit. O Radiation surveys are performed periodically for vacuum cleaners in use. O For areas in which a vacuum could become highly radioactive in a short period, remote monitoring is used. O Controls are in place to ensure that liquids are not vacuumed with units that are not designed for wet materials. O Physical controls such as a room, tent or containment bag are used to control the spread of contamination when contaminated vacuums are opened. O Breathing zone air is sampled each time a vacuum is opened. O All vacuum cleaner and hose openings are securely covered to prevent the spread of contamination. 4) Instrumentation Calibrate contamination survey equipment prior to initial use, at least annually, following repairs and whenever malfunction is known or suspected. At least each day an instrument (such as a whole-body contamination monitor, handheld frisker and gamma tool monitor) is in use to monitor personnel or equipment contamination, perform a response check using a radioactive source. For exceptions to daily response checks, see subsection C.2.a, Automatic contamination monitors. If more than one detector or alarm circuit may be used, then response check each detector or alarm circuit. The radioactive source used to check the alarm set points should have energy levels consistent with the station radionuclide mix and the strength of the source should provide confidence that the monitors will alarm at a level of 5,000 dpm/100 cm2 (beta and gamma).
Macro-Gap	SF15-05-15
Issue/Gap Description	There are instances of unclear standards in the RP Program, and current

	RP practices are not always documented in RP Program governance. BP-RPP-00045 does not have a mandatory requirement that all HEPA unit, vacuum cleaner, and hose openings be securely covered to prevent the spread of contamination when HEPA units or vacuums are not in use.
Rationale	This gap is an enhancement to the RP Program, not safety significant and can be dealt with through the Corrective Action process. The procedure is currently under revision and the gap has been dispositioned in the review. However, an AR will be entered to ensure a mandatory requirement in BP-RPP-00045 that all HEPA unit, vacuum cleaner, and hose openings be securely covered to prevent the spread of contamination when HEPA units or vacuums are not in use. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_VII.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VII.C2.
Requirement Assessed	a. Work procedures 1) Planning Sufficient preparation time is important when radiological work is being planned. Proper planning ensures a job will have controls in place to conduct work safely. Planning of the radiological aspects of work is integrated into the station work planning process and is the responsibility of job-planning personnel in conjunction with work group supervision and radiological protection personnel. Methods available for radiological control-such as engineered controls, shielding, efficiency improvement, decontamination, containment devices, work rescheduling, hot-spot flushing and mock-ups-are made a part of the job. When additional measures are not feasible for urgent jobs, sufficient management review should ensure that appropriate radiological controls are in place. Additionally, the work is evaluated to identify and document needed improvements for future jobs. During outages, the radiological protection organisation is actively involved with the implementation of outage plans and any decisions to deviate from those plans that may have a radiological impact. Radiological protection personnel monitor the outage schedule and emergent work to anticipate the need for and to plan radiological protection activities, minimising their impact on outage tasks while reducing collective dose. For emergent work activities, radiological protection, station management, and, as appropriate, the station ALARA Committee should ensure that appropriate additional controls and reviews are performed, to include effective planning and implementation of the work. Radiological protection personnel receive training on how to read and interpret the outage schedule and status reports. Changes in scheduled jobs are communicated to radiological protection technicians in the field. 2) Procedure use Plant operating and maintenance procedures or work documents for activities in elevated dose rate areas that involve significant collective dose, high contamination or the potential for the release of radioactive material include the important radiological protection actions identified during the planning process. This should include the requirement to notify Radiological Protection before these activities are initiated. Integrate radiological protection requirements into plant operation and maintenance procedures, whenever applicable, including action steps, hold points, notes, cautions and precautions. Radiological protection management reviews and concurs with procedures for activities with high radiation dose rates, accumulation of significant collective dose, high contamination or the potential for the release of radioactive material. Management establishes the expectation that personnel using these procedures will

comply with all required actions. Radiological protection supervision reviews procedure changes if the changes affect radiological protection requirements or radiological conditions for the work area.

b. Radiation work permits
Radiation work permits (RWPs) represent one of the primary administrative controls by which radiological work is planned and radiological control is implemented. In addition, they provide a means to trend radiation dose by specific jobs and to plan similar jobs in the future. The RWP is a formal, documented mechanism for radiological protection supervision to communicate radiological conditions and job controls to radiation workers. Involvement and accountability of all workers is part of the RWP implementation process.

1) General radiation work permits
Use general RWPs, or an equivalent administrative control, to govern routine work such as plant inspections, operator rounds and radiological protection technician surveys within the RCA. Radiological conditions for areas covered by general RWPs should be static, or the RWPs should address situations that could cause conditions to change. Clearly outline the type of work allowed under general RWPs for all radiation workers. Review routine surveys in areas covered by general RWPs for evidence of changing radiological conditions and revise the general RWP when appropriate.
Use general RWPs to control specific maintenance jobs only when approved by radiological protection supervision and when such jobs do not involve work with complex radiological conditions. General RWPs are not normally used for personnel entry into areas with dose rates of 1 mSv/hour at 30 cm or greater.

2) Specific radiation work permits
Use specific RWPs to control work in the RCA that is not covered by general RWPs. Such permits remain in effect only for the time needed to complete the job. Specific RWPs for jobs that are scheduled on a periodic basis, such as quarterly containment entries at power, are updated with current information before use and are only available for use by workers during the time scheduled for job performance. Perform surveys when radiological conditions are subject to change during the work, and revise the RWP as appropriate.
The following are examples for which specific RWPs are used to control work:
o Expected dose per worker exceeds 1 mSv.
o Dose rates are greater than 1 mSv/hour at 30 cm.
o Contamination levels of >=100,000 dpm/100 cm2 are involved or anticipated.
O Work is in an alpha level 3 area.
O A radiological hold point is necessary during the job (for example, a system breach).
O Work is done in airborne contamination areas.
O Radiological conditions are unknown.

	O Radiography operations are being conducted. O Protective clothing, special dosimetry or other requirements are needed that differ from standard requirements contained in general RWPs Specific job or task dose, dose accrual rate and work duration are desired for use by ALARA and job supervisors to support the capture of lessons learned and future performance improvement. 3) Radiation work permit preparation, approval and issuance Normally, the radiological protection organisation prepares, approves, issues and enforces RWPs and ALARA plans in accordance with written procedures. Steps include the following: o The job supervisor identifies all job activities and evolutions that could affect worker radiological protection. Maintenance requests and work procedures that govern the job, or other records such as an RWP request, are submitted to the radiological protection organisation to ensure complete understanding of the job. The radiological protection organisation receives this information with sufficient time to complete necessary radiological protection tasks prior to the planned work. These tasks may include determining radiological conditions, determining dose and contamination reduction actions, writing and approving RWPs, setting up the work area and scheduling radiological protection technician coverage. Specific RWPs include a clear, detailed description of the job location and the work to be performed. O Review previous job history as well as station and industry operating experience to evaluate and incorporate lessons learned. O Survey the work area for radiation, loose surface contamination, discrete radioactive particles and airborne radioactivity levels, as applicable for each type of radiation that presents a hazard to the worker (alpha, beta, gamma and neutron). This survey should identify the work area, contact and general area dose rates in and near the job location, including hot spots. The identification of low dose rate areas will assist workers in reducing their own dose. Document this information on the RWP. O Only rely on existing survey records in lieu of performing a new survey if they are current (that is, reflect present conditions) and appropriate (that is, include data on the types of radiation and the nature of contamination for the locations associated with the job). O Identify situations that require radiological hold points (for example, the work area cannot be completely surveyed because a system is not yet open). O Estimate person-rem for the job and determine appropriate dose reduction methods (if not already done). O Determine appropriate dose, contamination and solid radioactive waste controls. This should include the protective clothing, engineered controls, respirators, face shields, dosimeters and radiological hold points and the extent of radiological protection coverage needed for the job. When high radiation, discrete radioactive particle or airborne radioactivity conditions may be encountered, specify stop-work control levels (for example, dose rate or airborne radioactivity level) in the RWP at which all workers, including radiological protection personnel, are to leave the work

	area. O Compare the postings and boundary layout at the job site against that desired for the job and make necessary changes prior to the work. Expand contamination boundaries to provide enough space for workers to accomplish the job without inadvertently crossing the boundary. O Determine the appropriate level of radiological protection surveillance (job coverage). Include the use of remote cameras, audio communications and teledosimetry, when appropriate. O Determine the frequency and type of radiological surveys required during the job. O Determine the maximum allowed stay time in the area when there is a potential for high radiation exposure (for example, exposure rates greater than 15 mSv/hour at 30 cm or exposure greater than 5 mSv per entry) and how it will be monitored and enforced. Radiation work permits are approved by the appropriate level of designated radiological protection personnel. For example, the radiological protection manager approves entry into areas of 0.1 Sv/hour at 30 cm or above. Changes to RWPs require the same level of review and approval as the original. Prior to using an RWP, workers document that they have read the RWP, fully understand all requirements and radiological conditions, and agree to comply with these requirements. If conditions are not fully known when the RWP is issued, protective requirements are based on the best information available, with consideration of the most complex radiological conditions deemed probable. Therefore, avoid the use of a qualifier such as "as per Radiation Protection" for protective clothing or respirator requirements, because such qualifiers prevent workers from resolving questions and acknowledging that the instructions are understood. If conditions are not fully known when the RWP is issued, they should be determined as soon as possible or verified at the start of the work. Additionally, the probability of deficient protective requirements being prescribed at the job site increases because of insufficient forethought, work condition knowledge and planning and supervisory review.
Macro-Gap	SF15-01-15
Issue/Gap Description	The difference between general and specific REPs is not clearly described in the RP Program documentation.
Rationale	This gap is an enhancement to the RP Program, not safety significant and can be dealt with through the Corrective Action process. The procedure is currently under revision and the gap has been dispositioned in the review. An AR will be entered to ensure the difference between general and specific REPs is clearly described in the RP Program documentation This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Gap #	SF15_WANO GL 2004-01-R1_VII.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VII.C2.
Requirement Assessed	a. Work procedures 1) Planning Sufficient preparation time is important when radiological work is being planned. Proper planning ensures a job will have controls in place to conduct work safely. Planning of the radiological aspects of work is integrated into the station work planning process and is the responsibility of job-planning personnel in conjunction with work group supervision and radiological protection personnel. Methods available for radiological control-such as engineered controls, shielding, efficiency improvement, decontamination, containment devices, work rescheduling, hot-spot flushing and mock-ups-are made a part of the job. When additional measures are not feasible for urgent jobs, sufficient management review should ensure that appropriate radiological controls are in place. Additionally, the work is evaluated to identify and document needed improvements for future jobs. During outages, the radiological protection organisation is actively involved with the implementation of outage plans and any decisions to deviate from those plans that may have a radiological impact. Radiological protection personnel monitor the outage schedule and emergent work to anticipate the need for and to plan radiological protection activities, minimising their impact on outage tasks while reducing collective dose. For emergent work activities, radiological protection, station management, and, as appropriate, the station ALARA Committee should ensure that appropriate additional controls and reviews are performed, to include effective planning and implementation of the work. Radiological protection personnel receive training on how to read and interpret the outage schedule and status reports. Changes in scheduled jobs are communicated to radiological protection technicians in the field. 2) Procedure use Plant operating and maintenance procedures or work documents for activities in elevated dose rate areas that involve significant collective dose, high contamination or the potential for the release of radioactive material include the important radiological protection actions identified during the planning process. This should include the requirement to notify Radiological Protection before these activities are initiated. Integrate radiological protection requirements into plant operation and maintenance procedures, whenever applicable, including action steps, hold points, notes, cautions and precautions. Radiological protection management reviews and concurs with procedures for activities with high radiation dose rates, accumulation of significant collective dose, high contamination or the potential for the release of radioactive material. Management establishes the expectation that personnel using these procedures will

	comply with all required actions. Radiological protection supervision reviews procedure changes if the changes affect radiological protection requirements or radiological conditions for the work area. b. Radiation work permits Radiation work permits (RWPs) represent one of the primary administrative controls by which radiological work is planned and radiological control is implemented. In addition, they provide a means to trend radiation dose by specific jobs and to plan similar jobs in the future. The RWP is a formal, documented mechanism for radiological protection supervision to communicate radiological conditions and job controls to radiation workers. Involvement and accountability of all workers is part of the RWP implementation process. 1) General radiation work permits Use general RWPs, or an equivalent administrative control, to govern routine work such as plant inspections, operator rounds and radiological protection technician surveys within the RCA. Radiological conditions for areas covered by general RWPs should be static, or the RWPs should address situations that could cause conditions to change. Clearly outline the type of work allowed under general RWPs for all radiation workers. Review routine surveys in areas covered by general RWPs for evidence of changing radiological conditions and revise the general RWP when appropriate. Use general RWPs to control specific maintenance jobs only when approved by radiological protection supervision and when such jobs do not involve work with complex radiological conditions. General RWPs are not normally used for personnel entry into areas with dose rates of 1 mSv/hour at 30 cm or greater. 2) Specific radiation work permits Use specific RWPs to control work in the RCA that is not covered by general RWPs. Such permits remain in effect only for the time needed to complete the job. Specific RWPs for jobs that are scheduled on a periodic basis, such as quarterly containment entries at power, are updated with current information before use and are only available for use by workers during the time scheduled for job performance. Perform surveys when radiological conditions are subject to change during the work, and revise the RWP as appropriate. The following are examples for which specific RWPs are used to control work: o Expected dose per worker exceeds 1 mSv. o Dose rates are greater than 1 mSv/hour at 30 cm. o Contamination levels of >=100,000 dpm/100 cm2 are involved or anticipated. O Work is in an alpha level 3 area. O A radiological hold point is necessary during the job (for example, a system breach). O Work is done in airborne contamination areas. O Radiological conditions are unknown.

	O Radiography operations are being conducted. O Protective clothing, special dosimetry or other requirements are needed that differ from standard requirements contained in general RWPs Specific job or task dose, dose accrual rate and work duration are desired for use by ALARA and job supervisors to support the capture of lessons learned and future performance improvement. 3) Radiation work permit preparation, approval and issuance Normally, the radiological protection organisation prepares, approves, issues and enforces RWPs and ALARA plans in accordance with written procedures. Steps include the following: o The job supervisor identifies all job activities and evolutions that could affect worker radiological protection. Maintenance requests and work procedures that govern the job, or other records such as an RWP request, are submitted to the radiological protection organisation to ensure complete understanding of the job. The radiological protection organisation receives this information with sufficient time to complete necessary radiological protection tasks prior to the planned work. These tasks may include determining radiological conditions, determining dose and contamination reduction actions, writing and approving RWPs, setting up the work area and scheduling radiological protection technician coverage. Specific RWPs include a clear, detailed description of the job location and the work to be performed. O Review previous job history as well as station and industry operating experience to evaluate and incorporate lessons learned. O Survey the work area for radiation, loose surface contamination, discrete radioactive particles and airborne radioactivity levels, as applicable for each type of radiation that presents a hazard to the worker (alpha, beta, gamma and neutron). This survey should identify the work area, contact and general area dose rates in and near the job location, including hot spots. The identification of low dose rate areas will assist workers in reducing their own dose. Document this information on the RWP. O Only rely on existing survey records in lieu of performing a new survey if they are current (that is, reflect present conditions) and appropriate (that is, include data on the types of radiation and the nature of contamination for the locations associated with the job). O Identify situations that require radiological hold points (for example, the work area cannot be completely surveyed because a system is not yet open). O Estimate person-rem for the job and determine appropriate dose reduction methods (if not already done). O Determine appropriate dose, contamination and solid radioactive waste controls. This should include the protective clothing, engineered controls, respirators, face shields, dosimeters and radiological hold points and the extent of radiological protection coverage needed for the job. When high radiation, discrete radioactive particle or airborne radioactivity conditions may be encountered, specify stop-work control levels (for example, dose rate or airborne radioactivity level) in the RWP at which all workers, including radiological protection personnel, are to leave the work

	area. O Compare the postings and boundary layout at the job site against that desired for the job and make necessary changes prior to the work. Expand contamination boundaries to provide enough space for workers to accomplish the job without inadvertently crossing the boundary. O Determine the appropriate level of radiological protection surveillance (job coverage). Include the use of remote cameras, audio communications and teledosimetry, when appropriate. O Determine the frequency and type of radiological surveys required during the job. O Determine the maximum allowed stay time in the area when there is a potential for high radiation exposure (for example, exposure rates greater than 15 mSv/hour at 30 cm or exposure greater than 5 mSv per entry) and how it will be monitored and enforced. Radiation work permits are approved by the appropriate level of designated radiological protection personnel. For example, the radiological protection manager approves entry into areas of 0.1 Sv/hour at 30 cm or above. Changes to RWPs require the same level of review and approval as the original. Prior to using an RWP, workers document that they have read the RWP, fully understand all requirements and radiological conditions, and agree to comply with these requirements. If conditions are not fully known when the RWP is issued, protective requirements are based on the best information available, with consideration of the most complex radiological conditions deemed probable. Therefore, avoid the use of a qualifier such as "as per Radiation Protection" for protective clothing or respirator requirements, because such qualifiers prevent workers from resolving questions and acknowledging that the instructions are understood. If conditions are not fully known when the RWP is issued, they should be determined as soon as possible or verified at the start of the work. Additionally, the probability of deficient protective requirements being prescribed at the job site increases because of insufficient forethought, work condition knowledge and planning and supervisory review.
Macro-Gap	SF15-01-15
Issue/Gap Description	SEC-RPR-00015 Radiation Exposure Permit (REP) procedure does not explicitly require stop-work (or back-out) levels to be specified for DRPs or airborne particulates.
Rationale	This gap is an enhancement to the RP Program, not safety significant and can be dealt with through the Corrective Action process. The procedure is currently under revision and the gap has been dispositioned in the review. An AR will be entered to ensure SEC-RPR-00015 Radiation Exposure Permit (REP) procedure explicitly require stop-work (or back-out) levels to be specified for DRPs or airborne particulates. This gap is categorized as “Closed” in the database based on feedback from Bruce Power. Any follow-up actions or oversight is recorded in the database including associated ARs (Action Requests).

Table 51: Micro-gaps Identified by the CNSC with Safety Improvements Considered Unnecessary to Implement as Part of IIP

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-1_Comment 3
Document ID	REGDOC-2.5.2
Article/Clause	7.6.5.2
Requirement Assessed	SSCs important to safety shall typically not be shared between two or more reactors. In exceptional cases when SSCs are shared between two or more reactors, such sharing shall exclude safety systems and turbine generator buildings that contain high-pressure steam and feedwater systems, unless this contributes to enhanced safety.
Macro-Gap	n/a
Issue/Gap Description	Safety systems such as the ECCS and the Containment as well as the turbine building, are shared by the 4 reactor units. This clause explicitly states that such sharing shall exclude safety systems and turbine generator buildings.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-1_Comment 11
Document ID	SF2
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine the actual condition of Systems, Structures and Components (SSCs) important to safety and whether it is adequate for them to meet their design requirements. In addition, the review should confirm that the condition of SSCs is properly documented.
Macro-Gap	n/a
Issue/Gap Description	Known gaps in knowledge of the condition of SSCs were not reported in SFR-2. During the review of Bruce Power documents, some instances of unknown condition and missing data have been identified. The main objective of SFR-2 is to determine the actual condition of the SSCs important to safety. Where data is lacking, it should be generated or derived by performing plant walk downs, inspections, or tests at early stages of the PSR. Otherwise missing data and unknown conditions should be declared as gaps.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-1_Comment 27
Document ID	SF9
Article/Clause	RT
Requirement Assessed	Review the effectiveness of such programmes for the timely feedback of operating experience and for their output.
Macro-Gap	n/a
Issue/Gap Description	There are a high number of corrective actions that are deemed ineffective (-26%). Given that effectiveness reviews are only conducted on root causes rather than contributing causes of events, this seems high. While it is understood that there is a process for dealing with ineffective corrective actions, the cause of the issue should be addressed.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-1_Comment 30
Document ID	SF11
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine whether the operating organization's processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.
Macro-Gap	n/a
Issue/Gap Description	In many CNSC inspections referenced in different Safety Factor Reports (e.g. section 4.2 of implementation of the Engineering Change Control Process BRPD- AB-2013-011, section 4.4 of Restart Effectiveness inspection BRPD-A-2013-009, section 4.7 finding #2 of Problem Identification and Resolution Inspection BRPDAB- 2014-007) and in the recent inspection on Supply management program at Bruce Power BRPDAB-2015-001 as described by section 4.3, CNSC staff have found records that were not complete or with incorrect information. In many cases, the lack of attention to details was observed in the records. Bruce Power took corrective actions for these specific CNSC inspections but the problems should be addressed globally.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 2
Document ID	CSA N290.3-11
Article/Clause	4.1, 7.6, 9.4.1, 11.1.1
Requirement Assessed	This standard presents the requirements for the design, qualification, installation, operation, maintenance, inspection, and documentation of a containment system.
Macro-Gap	n/a
Issue/Gap Description	Compliance with N290.3-11 clause 4.1, clause 7.6, clause 9.4.1 and clause 11.1.1 has not been demonstrated. The capability of containment to cope with BDBAs including severe accidents in terms of preventing enhanced leakages and uncontrolled releases of radioactive materials has not been demonstrated. Therefore, the compliance category for these clauses should be "gap", though it is noted that Action Item 2015-07-3683 has been raised to address this item.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 3
Document ID	CSA N290.3-11
Article/Clause	7.4
Requirement Assessed	This standard presents the requirements for the design, qualification, installation, operation, maintenance, inspection, and documentation of a containment system.
Macro-Gap	n/a
Issue/Gap Description	N290.3-11 clause 7.4 states that: "Leakage limits shall be defined for both gas and liquid phases." Section A.12 of SFR1 (e-doc 5062285) discusses containment gas leakage, and it is missing compliance discussion regarding containment liquid phase's leakage. In addition, REGDOC-3.1.1 report B-2015-28496453 also notes that liquid releases are not considered in the Final Safety Analysis.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 4
Document ID	CSA N290.3-11
Article/Clause	9.5.6
Requirement Assessed	This standard presents the requirements for the design, qualification, installation, operation, maintenance, inspection, and documentation of a containment system.
Macro-Gap	n/a
Issue/Gap Description	N290.3-11 clause 9.5.6 states that:" In the case of severe accidents, emergency venting of containment shall consider the buildup of combustible gases to minimize injury to plant personnel and damage to SSCs from deflagration of combustible mixtures. Bruce Power has submitted JP 4426 documents related to the generic technical basis for in-vessel retention. However, there is no submission on Bruce Power A/B specific calandria vessel integrity assessment for severe accident conditions. The success of in-vessel retention implementation and the restoration of an active containment heat sink during a severe accident are important factors for evaluating the need for CFVS. It is noted that this subject continues to be discussed as part of the BDBAs for the station.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 5
Document ID	REGDOC-2.5.2
Article/Clause	8.4.1
Requirement Assessed	… For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all AOOs and DBAs in time to meet the respective derived acceptance criteria."
Macro-Gap	n/a
Issue/Gap Description	Bruce Power indicated that the assessment of clause 8.4.1 in REGDOC-2.5.2 was a gap for Bruce A but not for Bruce B. They designs are equivalent, so this should have been identified as a gap for Bruce B as well.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 6
Document ID	REGDOC-2.5.2
Article/Clause	7.6
Requirement Assessed	… The safety systems and their support systems shall be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 1E-3. …
Macro-Gap	n/a
Issue/Gap Description	REGDOC-2.5.2 clause 7.6 requires that special safety systems, including their support systems, meet a 10-3 availability target. Bruce Power’s Safety Factor Report did not indicate that the 10-3 target is met when considering the support systems and clarification did not address this point.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 12
Document ID	REGDOC-2.5.2
Article/Clause	7.1
Requirement Assessed	The design authority shall classify SSCs using a consistent and clearly defined classification method. The SSCs shall then be designed, constructed, and maintained such that their quality and reliability is commensurate with this classification. In addition, all SSCs shall be identified as either important or not important to safety. … SSCs important to safety shall include: 1. safety systems 2. complementary design features 3. safety support systems 4. other SSCs whose failure may lead to safety concerns (e.g., process and control systems)
Macro-Gap	n/a
Issue/Gap Description	At this time, Bruce Power does not have a comprehensive and ranked list of "Systems Importance to Safety" as per the definition and methodology of REGDOC-2.5.2. Document BP-PROC-00169 is almost complete and may eventually meet the requirements of REGDOC-2.5.2 once completed and SSCs are ranked. A list produced from DPT-RS-00012 cannot fulfill the requirements of REGDOC-2.5.2. DPT-RS-00012 is only PSA based and was solely produce for the reliability program. This is why it is a very small subset (less than 10%) of the more general list mandated by section 7.1 of REGDOC-2.5.2 as a governing list for all programs, including programs for pressure boundary, reliability, maintenance, reporting, EQ, QA, asset management, LCMPs, etc. BP-PROC-00169 and DPT-RS-00012 (even if combined) are not meeting REGDOC-2.5.2 requirements and this is considered to be a gap.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 15
Document ID	SF2
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine the actual condition of Systems, Structures and Components (SSCs) important to safety and whether it is adequate for them to meet their design requirements. In addition, the review should confirm that the condition of SSCs is properly documented.
Macro-Gap	n/a
Issue/Gap Description	In Sections 7.1.1.2, 7.1.1.4 and 7.1.1.5 of the report, Bruce Power indicates that there are SCRs that remain open following FASA of the Asset Management Program, Inspection Services Department Governance and Pipe Support Inspection Scope and Resourcing reviews. Specifically, this includes 28477152, 28504163, 28504168, 28525689 and 28525691. This indicates that there are gaps in the program that need to be resolved and long as corrective actions have not been completed.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 19
Document ID	SF2
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine the actual condition of Systems, Structures and Components (SSCs) important to safety and whether it is adequate for them to meet their design requirements. In addition, the review should confirm that the condition of SSCs is properly documented.
Macro-Gap	n/a
Issue/Gap Description	In section 7.1.1.6 of SFR-2 Bruce Power indicated that the review of the Relief Valve Quality Evaluation Program identified a number of opportunities for improvement. One corrective action was to update the next edition of the RV Quality Program Manual to prevent deferral of RV testing (relief valve testing being deferred is a gap against ASME OM Code and thus against CSA N285.0). This is considered a programmatic gap until the RV Quality Program Manual is updated.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 27
Document ID	REGDOC-2.4.1
Article/Clause	3
Requirement Assessed	The objectives of deterministic analysis are to: … 3. assist in establishing and validating accident management procedures and guidelines
Macro-Gap	n/a
Issue/Gap Description	Clause 3 (Section 3) of REGDOC-2.4.1 states that “The objectives of deterministic analysis are to 1).., 2)…, 3) assist in establishing and validating accident management procedures and guidelines, and 4) …”. Bruce Power’s compliance assessment against objective 3 states that “Accident management procedures are documented in the operating manuals”. There is no mention of severe accident management guidelines (SAMG) and no compliance discussion on how deterministic analysis is used to “assist in establishing and validating” the station-specific SAMG. Bruce Power has implied in this safety factor assessment that the technical basis of the generic SAMG has been confirmed by MAAP4-CANDU simulations and some deterministic severe accident analyses were performed under the framework of Level 2 PRA to support safety goal evaluations. However, it is recognized that there is lack of deterministic analyses for the purpose to support the implementation and evaluation of the station-specific SAMG. For example, structural and thermal-stress analyses of the Bruce B calandria vessel under static and dynamic loading conditions expected from a severe accident have not been performed to support the in-vessel retention (IRV) strategy and thus to support Bruce Power’s position that there is no need for installation of a severe accident containment filtered venting system (CFVS). Furthermore, the positive and negative impacts of various key mitigating actions as specified in SAMG are only qualitatively evaluated or judged in the SAMG, but not confirmed by deterministic analysis. Therefore, CNSC staff considers that a gap exists for Bruce Power to comply with objective 3 given in Section 3 of REGDOC-2.4.1.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 34
Document ID	SF10
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine whether the organization and administration are adequate for the safe operation of the nuclear power plant.
Macro-Gap	n/a
Issue/Gap Description	Despite improvements in performing the assessments for different programs there were no assessments (audit and self-assessments) for key programs in the three year period as per internal governance. For example, Safety Factor Report 6 'Probabilistic Safety Assessments' states that that there were no assessments (audits and self-assessments) performed for PRA. The effectiveness of the programs for PRA was not determined. This was raised in CNSC Inspection Report BRPD-AB-2014-004, however the action item was closed, but no corrective actions appear to have been implemented.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 35
Document ID	SF10
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine whether the organization and administration are adequate for the safe operation of the nuclear power plant.
Macro-Gap	n/a
Issue/Gap Description	In many CNSC inspections referenced in different Safety Factor Reports (e.g. section 4.2 of Implementation of the Engineering Change Control Process BRPDAB-2013-011, section 4.7 finding #2 of Problem Identification and Resolution Inspection BRPD-AB-2014-007) and in the recent inspection on Supply management program at Bruce Power BRPD-AB-2015-001 as described by section 4.3, CNSC staff have found records that were not complete or with incorrect information. In many cases, the lack of attention to details was observed in the records. Bruce Power took corrective actions for these specific CNSC inspections but the issues should be addressed globally.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 36
Document ID	SF10
Article/Clause	RT
Requirement Assessed	The specific objective of the review of this Safety Factor is to determine whether the organization and administration are adequate for the safe operation of the nuclear power plant.
Macro-Gap	n/a
Issue/Gap Description	There appear to be systemic issues with BP-PROC-00019 Action tracking as the process is not effective as documented and implemented. Section 7.2.13 AU-2012-00016 Procurement Engineering states that Action Tracking assignments were not completed when the document were revised. Although actions were implemented, CNSC staff found a few Action Requests (ARs) that were still not completed when procedures were already revised and due date for AR completion was not followed (i.e. no completion notes issued). The examples are: 1. AR 28264775 is for BP-PROC-00363 R003 that was issued in 2013 and still open. 2. AR 28503941 is noted in section 7.1.2 of SF 10 and the report stated that it was completed but AR is still open. 3. AR28389131 for changes to the procedures and issued over a long time ago (more than 3 years as document change cycle is defined in BP-PROG-03.01) is still open. 4. CNSC staff observed that AR28486169, 28484880, 28484876 completion notes were not competed as per due date in 2015. In addition, CNSC staff noted that section 4.12 of procedure BP-PROC-00019 Action Tracking R010 states that all assignments must have completion notes recorded. In accordance with section 4.19, “Cognos reports for Action taking oversight are maintained by the Performance Improvement Department for use by all Alert Group owners at their discretion”. However, the procedure does not provide the confidence in an effectiveness oversight process AR process. The verification of these Cognos reports is not mandatory (i.e. at Alert Group Owners discretion) and there is no direction (i.e., criteria were not specified) whether the Cognos reports will be followed for actions. These systemic issues should be considered a gap.
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 43
Document ID	REGDOC-2.3.2
Article/Clause	3.2
Requirement Assessed	Licensees shall: 6. conduct periodic reviews, drills and integrated exercises to confirm or improve the effectiveness of the established accident management measures 7. ensure that the accident management processes and activities interface with the emergency preparedness
Macro-Gap	n/a

Issue/Gap Description	REGDOC 2.3.2, Accident Management, general requirement 3.2, clauses 6 and 7 should have been categorized as gaps instead of indirect compliance:
	
	• Clause 6, Licensee shall "conduct periodic reviews, drills and integrated exercises to confirm or improve the effectiveness of the established accident management measures":
	  o There is no indication on how often/periodic review period is for the accident management program.
	  o There is no validation process in place to confirm or improve the effectiveness of the established accident management measures (i.e. SAMGs) through periodic reviews and integrated exercises.
	  o In addition, full implementation of SAMGs (and not just the initiation of SAMGs) should be exercised and/or drilled to test the full spectrum of severe accident management actions. This should be conducted on a defined periodic basis.
	  o SAMG verification drills and exercises schedule should be developed with the associated objectives and scope on an annual basis, similar to the EP drills schedule.
	• Clause 7, Licensee shall "ensure that the accident management processes and activities interface with emergency preparedness":
	  o There is no clear description (plan and procedures) in place that defines how the accident management processes and activities interface with emergency preparedness.
	
	It is noted that this is being addressed through Fukushima Action Items 3.1.1, 3.1.2, 3.1.3 and 3.1.4.

Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Gap #	eDoc 5229600_(NK21-CORR-00531-13581)_Table C-2_Comment 44
Document ID	SF15
Article/Clause	RT
Requirement Assessed	The review of RP equipment and instrumentation is described by REGDOC-2.3.3 as intended to demonstrate that there are adequate provisions "for monitoring all significant radiation sources, in all activities throughout the lifetime of the reactor facility. These should cover operational states and accident conditions and, as practicable, beyond-design-basis accidents, including severe accidents" (Section A.3.2).
Macro-Gap	n/a
Issue/Gap Description	Bruce Power's reviews did not address the aging and obsolescence of fixed RP instrumentation (e.g. Fixed Area Gamma monitors and Tritium Area Monitors) that could impact radiological safety. The review did not also include the review of the physical condition of RP instrumentation and equipment that should be confirmed by walk downs where practicable to verify continued utility and functionality. The additional information requests raised by CNSC staff were not provided to clarify whether this RP instrumentation remains fit for service, or will require replacement within the timeframe covered by this PSR. As a result, a gap was identified to address the issue of ageing and obsolescence of fixed RP instrumentation (e.g. Fixed Area Gamma monitors and Tritium Area Monitors).
Rationale	This is a CNSC-raised gap that will be addressed in a manner analogous to the Category 2 gaps, in that Bruce Power will establish an AR number to track how each is being addressed.

Appendix F – CATEGORY 3: Safety Improvement In-Progress

Appendix F consists of those micro-gaps identified in the Safety Factor Reports for which safety improvements are in progress.

- Table 52 provides a consolidation of all micro-gaps within this category. It is ordered such that gaps that are similar or identical appear consecutively. This can be regarded as a "smart table of contents" for the micro-gaps discussed in the next bullet, and provides a direct linkage back to the origin of the micro-gaps in the Safety Factor Reports.
- Table 53 provides the details for each of the micro-gaps within this category. This is based on an export from the PSR database, and is ordered first by Safety Factor, then by regulatory document/code/standard, then by clause.

The micro-gap number, which is provided in both tables, facilitates their use.

Table 52: Consolidation of Micro-gaps for Which Safety Improvements are in Progress

Category 3- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_CNSC REGDOC 2.5.2_4.2.1_15	SF01-01-15	CNSC REGDOC 2.5.2	4.2.1	5
SF01_CNSC REGDOC 2.5.2_4.2.1_16	SF01-01-16	CNSC REGDOC 2.5.2	4.2.1	5
SF01_CNSC REGDOC 2.5.2_4.2.3_16	SF01-01-16	CNSC REGDOC 2.5.2	4.2.3	7
SF01_CNSC REGDOC 2.5.2_4.2.3_15	SF01-03-15	CNSC REGDOC 2.5.2	4.2.3	7
SF01_CNSC REGDOC 2.5.2_7.3.4_16	SF01-01-16	CNSC REGDOC 2.5.2	7.3.4	7
SF01_CNSC REGDOC 2.5.2_7.3.4_15	SF01-01-15	CNSC REGDOC 2.5.2	7.3.4	7
SF01_CNSC REGDOC 2.5.2_7.4_16	SF01-01-16	CNSC REGDOC 2.5.2	7.4	7
SF01_CNSC REGDOC 2.5.2_7.4_15	SF01-01-15	CNSC REGDOC 2.5.2	7.4	7
SF01_CNSC REGDOC 2.5.2_9.1_16	SF01-01-16	CNSC REGDOC 2.5.2	9.1	7
SF05_CNSC REGDOC 2.4.1_4.2.3_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.3	7
SF05_CNSC REGDOC 2.4.1_4.2.3_15	SF05-01-15	CNSC REGDOC 2.4.1	4.2.3	7
SF05_CNSC REGDOC 2.5.2_7.4_15	SF05-02-15	CNSC REGDOC 2.5.2	7.4	7
SF05_CNSC REGDOC 2.5.2_7.4_16	SF05-02-16	CNSC REGDOC 2.5.2	7.4	7
SF05_CNSC REGDOC 2.5.2_9.1_15	SF05-02-15	CNSC REGDOC 2.5.2	9.1	7
SF05_CNSC REGDOC 2.5.2_9.1_16	SF05-02-16	CNSC REGDOC 2.5.2	9.1	7
SF05_CNSC REGDOC 2.5.2_9.4_16	SF05-02-16	CNSC REGDOC 2.5.2	9.4	7
SF05_CNSC REGDOC 2.5.2_9.4_15	SF05-02-15	CNSC REGDOC 2.5.2	9.4	7
SF01_CNSC REGDOC 2.5.2_4.2.3_16	SF01-03-16	CNSC REGDOC 2.5.2	4.2.3	8
SF01_CNSC REGDOC 2.5.2_4.2.3_15	SF01-01-15	CNSC REGDOC 2.5.2	4.2.3	8
SF01_CNSC REGDOC 2.5.2_5.3_16	SF01-04-16	CNSC REGDOC 2.5.2	5.3	9

Category 3- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_CNSC REGDOC 2.5.2_5.3_15	SF01-04-15	CNSC REGDOC 2.5.2	5.3	9
SF01_CNSC REGDOC 2.5.2_6.1_16	SF01-01-16	CNSC REGDOC 2.5.2	6.1	10
SF01_CNSC REGDOC 2.5.2_6.1_15	SF01-01-15	CNSC REGDOC 2.5.2	6.1	10
SF01_CNSC REGDOC 2.5.2_6.6.1_16	SF01-03-16	CNSC REGDOC 2.5.2	6.6.1	13
SF01_CNSC REGDOC 2.5.2_6.6.1_15	SF01-03-15	CNSC REGDOC 2.5.2	6.6.1	13
SF01_CNSC REGDOC 2.5.2_7.3_15	SF01-01-15	CNSC REGDOC 2.5.2	7.3	14
SF01_CNSC REGDOC 2.5.2_7.3_16	SF01-01-16	CNSC REGDOC 2.5.2	7.3	14
SF01_CNSC REGDOC 2.5.2_7.7_16	SF01-06-16	CNSC REGDOC 2.5.2	7.7	22
SF01_CNSC REGDOC 2.5.2_7.7_15	SF01-06-15	CNSC REGDOC 2.5.2	7.7	22
SF01_CNSC REGDOC 2.5.2_7.15.3_16	SF01-10-16	CNSC REGDOC 2.5.2	7.15.3	31
SF01_CNSC REGDOC 2.5.2_7.15.3_15	SF01-10-15	CNSC REGDOC 2.5.2	7.15.3	31
SF01_CNSC REGDOC 2.5.2_8.1.1_15	SF01-01-15	CNSC REGDOC 2.5.2	8.1.1	33
SF01_CNSC REGDOC 2.5.2_8.1.1_16	SF01-01-16	CNSC REGDOC 2.5.2	8.1.1	33
SF01_CNSC REGDOC 2.5.2_8.4.1_15	SF01-01-15	CNSC REGDOC 2.5.2	8.4.1	35
SF01_CNSC REGDOC 2.5.2_8.4.1_16	SF01-01-16	CNSC REGDOC 2.5.2	8.4.1	35
SF01_CNSC REGDOC 2.5.2_8.9.3_16	SF01-12-16	CNSC REGDOC 2.5.2	8.9.3	42
SF01_CNSC REGDOC 2.5.2_8.9.3_15	SF01-12-15	CNSC REGDOC 2.5.2	8.9.3	42
SF01_CNSC REGDOC 2.5.2_9.1_15	SF01-13-15	CNSC REGDOC 2.5.2	9.1	51
SF01_CNSC REGDOC 2.5.2_9.1_16	SF01-13-16	CNSC REGDOC 2.5.2	9.1	51
SF01_CNSC REGDOC 2.5.2_9.2_16	SF01-01-16	CNSC REGDOC 2.5.2	9.2	52
SF01_CNSC REGDOC 2.5.2_9.2_15	SF01-01-15	CNSC REGDOC 2.5.2	9.2	52
SF01_CNSC REGDOC 2.5.2_9.3_15	SF01-09-15	CNSC REGDOC 2.5.2	9.3	53
SF01_CSA N290.1_4.3.1.4_16	SF01-09-16	CSA N290.1	4.3.1.4	61
SF05_CSA N290.1_4.3.1.4_16	SF05-09-16	CSA N290.1	4.3.1.4	61
SF04_CSA N285.4-14_12.5_16	SF04-02-16	CSA N285.4-14	12.5	77
SF04_CSA N285.4-14_12.5_15	SF04-02-15	CSA N285.4-14	12.5	77
SF05_CNSC REGDOC 2.3.2_3.4_15	SF05-06-15	CNSC REGDOC 2.3.2	3.4	78
SF05_CNSC REGDOC 2.3.2_4.2.1_15	SF05-06-15	CNSC REGDOC 2.3.2	4.2.1	78
SF05_CNSC REGDOC 2.3.2_4.2.5_15	SF05-06-15	CNSC REGDOC 2.3.2	4.2.5	78
SF05_CNSC REGDOC 2.4.1_3_15	SF05-11-15	CNSC REGDOC 2.4.1	3	79
SF05_CNSC REGDOC 2.4.1_3_15	SF05-01-15	CNSC REGDOC 2.4.1	3	80
SF05_CNSC REGDOC 2.4.1_4.1_15	SF05-01-15	CNSC REGDOC 2.4.1	4.1	81
SF05_CNSC REGDOC 2.4.1_4.1_16	SF05-01-16	CNSC REGDOC 2.4.1	4.1	81
SF05_CNSC REGDOC 2.4.1_4.4.2_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.2	81
SF05_CNSC REGDOC 2.4.1_4.4.2_16	SF05-08-16	CNSC REGDOC 2.4.1	4.4.2	81
SF05_CNSC REGDOC 2.5.2_9.4_15	SF05-01-15	CNSC REGDOC 2.5.2	9.4	81
SF05_CNSC REGDOC 2.5.2_9.4_16	SF05-01-16	CNSC REGDOC 2.5.2	9.4	81
SF05_CNSC REGDOC 2.4.1_4.2.1_15	SF05-02-15	CNSC REGDOC 2.4.1	4.2.1	84

Category 3- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF05_CNSC REGDOC 2.4.1_4.2.1_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.1	84
SF05_CNSC REGDOC 2.4.1_4.2.2_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.2	85
SF05_CNSC REGDOC 2.4.1_4.2.2_15	SF05-02-15	CNSC REGDOC 2.4.1	4.2.2	85
SF05_CNSC REGDOC 2.4.1_4.2.2_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.2	86
SF05_CNSC REGDOC 2.4.1_4.2.2_15	SF05-02-15	CNSC REGDOC 2.4.1	4.2.2	86
SF05_CNSC REGDOC 2.4.1_4.2.2_15	SF05-02-15	CNSC REGDOC 2.4.1	4.2.2	87
SF05_CNSC REGDOC 2.4.1_4.2.3_16	SF05-08-16	CNSC REGDOC 2.4.1	4.2.3	88
SF05_CNSC REGDOC 2.4.1_4.2.3_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.3	89
SF05_CNSC REGDOC 2.4.1_4.2.3_16	SF05-02-16	CNSC REGDOC 2.4.1	4.2.3	91
SF05_CNSC REGDOC 2.4.1_4.2.3_15	SF05-10-15	CNSC REGDOC 2.4.1	4.2.3	91
SF05_CNSC REGDOC 2.4.1_4.4.2_16	SF05-11-16	CNSC REGDOC 2.4.1	4.4.2	91
SF05_CNSC REGDOC 2.4.1_4.4.2_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.2	91
SF05_CNSC REGDOC 2.4.1_4.3_15	SF05-02-15	CNSC REGDOC 2.4.1	4.3	92
SF05_CNSC REGDOC 2.4.1_4.3_16	SF05-02-16	CNSC REGDOC 2.4.1	4.3	92
SF05_CNSC REGDOC 2.5.2_4.2.1_15	SF05-02-15	CNSC REGDOC 2.5.2	4.2.1	92
SF05_CNSC REGDOC 2.5.2_4.2.1_16	SF05-02-16	CNSC REGDOC 2.5.2	4.2.1	92
SF05_CNSC REGDOC 2.5.2_9.2_15	SF05-02-15	CNSC REGDOC 2.5.2	9.2	92
SF05_CNSC REGDOC 2.5.2_9.2_16	SF05-02-16	CNSC REGDOC 2.5.2	9.2	92
SF05_CNSC REGDOC 2.4.1_4.3.2_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.2	95
SF05_CNSC REGDOC 2.4.1_4.3.2_16	SF05-03-16	CNSC REGDOC 2.4.1	4.3.2	95
SF05_CNSC REGDOC 2.4.1_4.3.4_16	SF05-03-16	CNSC REGDOC 2.4.1	4.3.4	98
SF05_CNSC REGDOC 2.4.1_4.3.4_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.4	98
SF05_CNSC REGDOC 2.4.1_4.3.4_15	SF05-03-15	CNSC REGDOC 2.4.1	4.3.4	99
SF05_CNSC REGDOC 2.4.1_4.3.4_16	SF05-07-16	CNSC REGDOC 2.4.1	4.3.4	99
SF05_CNSC REGDOC 2.4.1_4.4.1_16	SF05-05-16	CNSC REGDOC 2.4.1	4.4.1	102
SF05_CNSC REGDOC 2.4.1_4.4.1_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.1	102
SF05_CNSC REGDOC 2.4.1_4.4.1_16	SF05-01-16	CNSC REGDOC 2.4.1	4.4.1	103
SF05_CNSC REGDOC 2.4.1_4.4.1_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.1	103
SF05_CNSC REGDOC 2.4.1_4.4.1_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.1	104
SF05_CNSC REGDOC 2.4.1_4.4.1_16	SF05-07-16	CNSC REGDOC 2.4.1	4.4.1	104
SF05_CNSC REGDOC 2.4.1_4.4.2_16	SF05-07-16	CNSC REGDOC 2.4.1	4.4.2	105
SF05_CNSC REGDOC 2.4.1_4.4.2_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.2	105
SF05_CNSC REGDOC 2.4.1_4.4.2_16	SF05-11-16	CNSC REGDOC 2.4.1	4.4.2	106
SF05_CNSC REGDOC 2.4.1_4.4.2_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.2	106
SF05_CNSC REGDOC 2.4.1_4.4.2_16	SF05-01-16	CNSC REGDOC 2.4.1	4.4.2	107
SF05_CNSC REGDOC 2.4.1_4.4.2_15	SF05-10-15	CNSC REGDOC 2.4.1	4.4.2	107
SF05_CNSC REGDOC 2.4.1_4.4.1_15		CNSC REGDOC 2.4.1	4.4.1	110
SF05_CNSC REGDOC 2.4.1_4.4.3_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.3	110

Category 3- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF05_CNSC REGDOC 2.4.1_4.4.3_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.3	111
SF05_CNSC REGDOC 2.4.1_4.4.3_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.3	112
SF05_CNSC REGDOC 2.4.1_4.4.3_16	SF05-07-16	CNSC REGDOC 2.4.1	4.4.3	112
SF05_CNSC REGDOC 2.4.1_4.4.3_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.3	113
SF05_CNSC REGDOC 2.4.1_4.4.4_15	SF05-06-15	CNSC REGDOC 2.4.1	4.4.4	114
SF05_CNSC REGDOC 2.4.1_4.4.4_16	SF05-11-16	CNSC REGDOC 2.4.1	4.4.4	114
SF05_CNSC REGDOC 2.4.1_4.4.4_15	SF05-04-15	CNSC REGDOC 2.4.1	4.4.4	115
SF05_CNSC REGDOC 2.4.1_4.4.4_15	SF05-04-15	CNSC REGDOC 2.4.1	4.4.4	116
SF05_CNSC REGDOC 2.4.1_4.4.4_16	SF05-05-16	CNSC REGDOC 2.4.1	4.4.4	116
SF05_CNSC REGDOC 2.4.1_4.4.6_16	SF05-04-16	CNSC REGDOC 2.4.1	4.4.6	116
SF05_CNSC REGDOC 2.4.1_4.4.4_15	SF05-05-15	CNSC REGDOC 2.4.1	4.4.4	117
SF05_CNSC REGDOC 2.4.1_4.4.4_16	SF05-04-16	CNSC REGDOC 2.4.1	4.4.4	117
SF05_CNSC REGDOC 2.4.1_4.4.4_15	SF05-12-15	CNSC REGDOC 2.4.1	4.4.4	118
SF05_CNSC REGDOC 2.4.1_4.4.4_15	SF05-04-15	CNSC REGDOC 2.4.1	4.4.4	119
SF05_CNSC REGDOC 2.4.1_4.4.5_16	SF05-01-16	CNSC REGDOC 2.4.1	4.4.5	122
SF05_CNSC REGDOC 2.4.1_4.4.5_15	SF05-01-15	CNSC REGDOC 2.4.1	4.4.5	122
SF05_CNSC REGDOC 2.4.1_4.5_15	SF05-01-15	CNSC REGDOC 2.4.1	4.5	124
SF05_CNSC REGDOC 2.4.1_4.5_16	SF05-01-16	CNSC REGDOC 2.4.1	4.5	124
SF05_CNSC REGDOC 2.4.1_4.6.2_15	SF05-08-15	CNSC REGDOC 2.4.1	4.6.2	125
SF05_CNSC REGDOC 2.4.1_4.7_15	SF05-01-15	CNSC REGDOC 2.4.1	4.7	126
SF05_CNSC REGDOC 2.4.1_4.7_16	SF05-01-16	CNSC REGDOC 2.4.1	4.7	126
SF05_CNSC REGDOC 2.5.2_4.2.3_16	SF05-02-16	CNSC REGDOC 2.5.2	4.2.3	127
SF05_CNSC REGDOC 2.5.2_4.2.3_15	SF05-02-15	CNSC REGDOC 2.5.2	4.2.3	127
SF05_CNSC REGDOC 2.5.2_6.1_16	SF05-02-16	CNSC REGDOC 2.5.2	6.1	128
SF05_CNSC REGDOC 2.5.2_6.1_15	SF05-02-15	CNSC REGDOC 2.5.2	6.1	128
SF05_CNSC REGDOC 2.5.2_6.4_16	SF05-02-16	CNSC REGDOC 2.5.2	6.4	130
SF05_CNSC REGDOC 2.5.2_6.4_15	SF05-02-15	CNSC REGDOC 2.5.2	6.4	130
SF05_CNSC REGDOC 2.5.2_6.4_15	SF05-02-15	CNSC REGDOC 2.5.2	6.4	131
SF05_CNSC REGDOC 2.5.2_6.6.1_15	SF05-02-15	CNSC REGDOC 2.5.2	6.6.1	132
SF05_CNSC REGDOC 2.5.2_6.6.1_16	SF05-02-16	CNSC REGDOC 2.5.2	6.6.1	132
SF05_CNSC REGDOC 2.5.2_7.4_16	SF05-02-16	CNSC REGDOC 2.5.2	7.4	132
SF05_CNSC REGDOC 2.5.2_7.6.2_16	SF05-04-16	CNSC REGDOC 2.5.2	7.6.2	133
SF05_CNSC REGDOC 2.5.2_7.6.2_15	SF05-04-15	CNSC REGDOC 2.5.2	7.6.2	133
SF05_CNSC REGDOC 2.5.2_8.4.1_15	SF05-03-15	CNSC REGDOC 2.5.2	8.4.1	134
SF05_CNSC REGDOC 2.5.2_8.4.1_16	SF05-03-16	CNSC REGDOC 2.5.2	8.4.1	134
SF05_CNSC REGDOC 2.5.2_9.4_16	SF05-07-16	CNSC REGDOC 2.5.2	9.4	135
SF05_CNSC REGDOC 2.5.2_9.4_15	SF05-03-15	CNSC REGDOC 2.5.2	9.4	135
SF08_SF8 RT_5.6_16	SF08-04-16	SF8 RT	5.6	142

Category 3- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF08_SF8 RT 2015_5.6_15	SF08-05-15	SF8 RT 2015	5.6	142
SF08_SF8 RT 2015_5.3_15	SF08-03-15	SF8 RT 2015	5.3	149
SF08_SF8 RT 2015_5.5_15	SF08-08-15	SF8 RT 2015	5.5	150
SF11_SF11 RT_5.4_15	SF11-01-15	SF11 RT	5.4	150
SF08_SF8 RT 2015_5.7_15	SF08-06-15	SF8 RT 2015	5.7	151
SF12_CSA N290.12_4.1.2_16	SF12-05-16	CSA N290.12	4.1.2	163
SF12_CSA N290.12_4.1.6_16	SF12-05-16	CSA N290.12	4.1.6	164
SF12_CSA N290.12_4.3_16	SF12-05-16	CSA N290.12	4.3	165
SF12_CSA N290.12_4.3_16	SF12-05-16	CSA N290.12	4.3	166
SF12_CSA N290.12_5.2.1_16	SF12-05-16	CSA N290.12	5.2.1	167
SF12_CSA N290.12_5.2.3_16	SF12-05-16	CSA N290.12	5.2.3	169
SF12_CSA N290.12_5.2.4_16	SF12-05-16	CSA N290.12	5.2.4	170
SF12_CSA N290.12_5.3.1_16	SF12-05-16	CSA N290.12	5.3.1	171
SF12_CSA N290.12_5.3.4_16	SF12-05-16	CSA N290.12	5.3.4	173
SF12_CSA N290.12_5.4.2_16	SF12-05-16	CSA N290.12	5.4.2	174
SF12_CSA N290.12_5.4.4_16	SF12-05-16	CSA N290.12	5.4.4	175
SF12_CSA N290.12_6.1.6_16	SF12-05-16	CSA N290.12	6.1.6	177
SF12_CSA N290.12_6.2.2_16	SF12-05-16	CSA N290.12	6.2.2	178
SF12_CSA N290.12_6.3.1_16	SF12-05-16	CSA N290.12	6.3.1	179
SF12_CSA N290.12_6.3.2_16	SF12-05-16	CSA N290.12	6.3.2	180
SF12_CSA N290.12_6.3.3_16	SF12-05-16	CSA N290.12	6.3.3	180
SF12_CSA N290.12_6.4.1_16	SF12-05-16	CSA N290.12	6.4.1	181
SF12_CSA N290.12_6.4.2_16	SF12-05-16	CSA N290.12	6.4.2	181
SF12_CSA N290.12_6.5.3_16	SF12-05-16	CSA N290.12	6.5.3	182
SF12_CSA N290.12_6.5.4_16	SF12-05-16	CSA N290.12	6.5.4	183
SF12_CSA N290.12_7.1_16	SF12-05-16	CSA N290.12	7.1	184
SF12_CSA N290.12_8.5_16	SF12-05-16	CSA N290.12	8.5	185
SF12_CSA N290.12_8.6_16	SF12-05-16	CSA N290.12	8.6	186
SF12_CSA N290.12_8.8_16	SF12-05-16	CSA N290.12	8.8	187
SF12_CSA N290.12_8.9_16	SF12-05-16	CSA N290.12	8.9	188
SF12_CSA N290.12_8.11_16	SF12-05-16	CSA N290.12	8.11	189
SF12_CSA N290.12_8.12_16	SF12-05-16	CSA N290.12	8.12	190
SF12_NUREG-0700_Part_I_15	SF12-02-15	NUREG-0700	Part_I	192
SF12_NUREG-0700_Part_I_16	SF12-04-16	NUREG-0700	Part_I	192
SF12_NUREG-0700_Part_II_4_15	SF12-02-15	NUREG-0700	Part_II_4	193
SF13_CNSC REGDOC 2.10.1_2.1_15	SF13-02-15	CNSC REGDOC 2.10.1	2.1	207
SF13_CNSC REGDOC 2.10.1_2.1_16	SF13-01-16	CNSC REGDOC	2.1	207

Category 3- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
		2.10.1		
SF13_CNSC REGDOC 2.10.1_2.3.4_15	SF13-01-15	CNSC REGDOC 2.10.1	2.3.4	211
SF13_CNSC REGDOC 2.3.2_3.3_15	SF13-02-15	CNSC REGDOC 2.3.2	3.3	212
SF13_CNSC REGDOC 2.3.2_3.4_15	SF13-02-15	CNSC REGDOC 2.3.2	3.4	213
SF13_CNSC REGDOC 2.3.2_3.5_15	SF13-02-15	CNSC REGDOC 2.3.2	3.5	214
SF13_CSA N1600-14_4.6.1_16	SF13-02-16	CSA N1600-14	4.6.1	218
SF13_CSA N1600-14_4.6.1_15	SF13-04-15	CSA N1600-14	4.6.1	218
SF13_CNSC REGDOC 2.10.1_2.2.3_15	SF13-01-15	CNSC REGDOC 2.10.1	2.2.3	256

Notes on Table 53:

1. This requirement is also covered in REGDOC-2.4.1.
2. Bruce Power has undertaken to evaluate the significance of gaps against REGDOC-2.4.1, and their importance to safety shall be established and applied on an as-needed basis, providing a means of prioritizing safety analysis to deliver the greatest safety benefit. This gap is classified as "In-Progress" based on the following correspondence:

   - Letter, F. Saunders to K. Lafreniere, "Action Item 090739: Safety Report Improvement Project- Regulatory Communication Plan", November 24, 2015, NK21-CORR-00531-12334 / NK29-CORR-00531-12767

     The cover letter provides the following information:

     “Bruce Power's Regulatory Communication Plan to update Part 3 of the Safety Reports for Bruce A and B and implement REGDOC-2.4.1, is provided in Attachment A. Tables A1 and A2 of Attachment A provide the schedule for planned submissions and meetings in support of the 2017 Safety Report submission.”

     Further context details are provided in Attachment A- Regulatory Communication Plan for the Safety Report Improvement (SRI) Project:

     “Deterministic Safety Analysis high-level requirements are specified in Licence Condition 4.1 of the Power Reactor Operating Licence (PROL) for the Bruce A and Bruce B Stations [A-Al] and more specific compliance verification criteria are given in the Section 4.1 of the companion Licence Conditions Handbook (LCH) [R-A2]. As part of the compliance verification criteria identified in the LCH, a new Canadian Nuclear Safety Commission (CNSC) Regulatory Document "Deterministic Safety Analysis", REGDOC-2.4.1 has been identified for compliance by December 31, 2017. The Regulatory Document outlines new requirements related to safety analysis events, operating modes, acceptance criteria, methods, documentation and review. Therefore, a three year Safety Report Improvement (SRI) Project, scheduled to be completed by December 31, 2017, is to upgrade Part 3 of the Safety Reports to add a Common Mode Failure (CMF) Appendix) and align the Safety Report (SR) Framework with REGDOC-2.4.1.

     Bruce Power agrees with the CNSC that full compliance with REGDOC-2.4.1 may not be possible or may not provide additional safety benefit beyond the current safety case [RA2]. A graded approach has been adopted to evaluate the significance of the gaps against REGDOC-2.4.1 [R-A3] [R-A4]. Following the project improvements and enhancements, Bruce Power will programmatically ensure that new safety analysis and assessments are consistent with REGDOC-2.4.1 through the implementation of the ongoing Safety Analysis Improvement Program (SAIP) responsible for future updates to the SRs.”

   - Letter from F. Saunders to M. Leblanc, Bruce Power: Requests and Supplemental Information for Licence Renewal dated November 28, 2014, NK21-CORR-00531-11715 / NK29-CORR-00531-12105 states the following in Section A-3 REGDOC-2.4.1 (2014)- Deterministic Safety Analysis Summary of Disposition Results and Transition Measures:

     "As described in Bruce Power’s PROL renewal applications (References A1 and A2),Bruce Power is in the process of implementing a Safety Report Improvement (SRI) initiative which includes updates to the Bruce Power Deterministic Safety Analysis (DSA) governance (programs, processes and procedures) to ensure safety analyses are geared toward becoming consistent with RD-310 and now REGDOC-2.4.1. The CNSC has been tracking these activities with CSNC [sic] Action Item 090739."

In 'Other Impacts and Related Information' of this Letter 'Impact Statement' Section states the following for Responsible Alert Group DPTNSAS:

"There are existing ARs to update Part 3 of the Bruce A Safety Report (28285163) and the Bruce B Safety Report (28275999), the due dates for which have been revised based on CNSC acceptance of the Safety Report Improvement Plan and RD-310 compliance (NK21-CORR-00531-11214 / NK29-CORR-00531-11261). This letter changes RD-310 to REGDOC-2.4.1 but the due dates for the RegOs remain the same per the accepted the SRI plan."

As indicated above, REGDOC-2.4.1 compliance is implemented under CNSC Action Item 090739 which is included in the Integrated Implementation Plan submitted to the CNSC (Letter from F. Saunders to K. Lafreniere, 'Integrated Implementation Plan for Bruce A, Bruce B and Center of Site in the Next Licence Period', dated October 31, 2014, NK21-CORR-00531-11567, NK29-CORR-00531-11950, NK37-CORR-00531-02288).

- Letter, K. Lafreniere to F. Saunders, "Action Item 090739: Acceptance of Safety Report Improvement Plan for Bruce A and B", March 25, 2014, e-Docs 4407612, NK21-CORR-00531-11214 / NK29-CORR-00531-11621.
- Letter, F. Saunders to K. Lafreniere, "Action Item 090739: Safety Report Improvement Plan for Bruce A and B", November 20, 2013, NK21-CORR-00531-10774 / NK29-CORR-00531-11155.

3. For greater context of the gaps to CSA N286.7-99 with respect to the use of legacy codes, Clause 5.1 of N286.7-99 states the following.

   “For those computer programs developed prior to the promulgation of this Standard (1999) and not changed thereafter, Clause 6 and Clause 11.2 do not apply. However, if such programs are used in performing substantial new safety or licensing analyses, the user organization shall prepare a plan. This plan shall

   (a) identify the extent to which the computer program conforms with the requirements of Clauses 6 and 11.2;

   (b) provide justification for nonconformance with Clauses 6 and 11.2;

   (c) define what verification activities will be performed and the verification needed; and

   (d) identify the time scale over which the verification activities will be performed.”

   Additionally, Bruce Power has undertaken to evaluate the significance of gaps against REGDOC-2.4.1 and their importance to safety shall be established and applied on an as-needed basis, providing a means of prioritizing safety analysis to deliver the greatest safety benefit. This will be the driver in determining what new analysis will be performed and hence achieve greater alignment with CSA N286.7-99 over the long term as per Bruce Power procedures.

Table 53: Micro-gaps with Safety Improvements In-Progress

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.1 Dose acceptance criteria
Requirement Assessed	The acceptance criteria for normal operations are provided in section 6.4. The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, shall be calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose shall be less than or equal to the dose acceptance criteria of: 1. 0.5 millisievert (mSv) for any AOO or 2. 20 mSv for any DBA The values adopted for the dose acceptance criteria for AOOs and DBAs are consistent with accepted international practices, and take into account the recommendations of the IAEA and the International Commission on Radiological Protection.
Macro-Gap	SF01-01-15
Issue/Gap Description	The Bruce A safety analysis covers a wide range of accident scenarios, demonstrating that the levels of defence-in-depth have been met, and that all of the regulatory reference dose limits of the current licence are not exceeded. However, the AOOs have not been analyzed explicitly to demonstrate that the specific dose acceptance criteria are met (Gap). It should be noted that although AOOs have not been directly addressed in the analysis, they have been shown to meet the current single failure limit, as required.
Rationale	See Notes 1 and 2s

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.1 Dose acceptance criteria
Requirement Assessed	The acceptance criteria for normal operations are provided in section 6.4. The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, shall be calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose shall be less than or equal to the dose acceptance criteria of: 1. 0.5 millisievert (mSv) for any AOO or 2. 20 mSv for any DBA The values adopted for the dose acceptance criteria for AOOs and DBAs are consistent with accepted international practices, and take into account the recommendations of the IAEA and the International Commission on Radiological Protection.
Macro-Gap	SF01-01-16
Issue/Gap Description	The Bruce B safety analysis covers a wide range of accident scenarios, demonstrating that the levels of defence-in-depth have been met, and that all of the regulatory reference dose limits of the current licence are not exceeded. However, the AOOs have not been analyzed explicitly to demonstrate that the specific dose acceptance criteria are met (Gap). It should be noted that although AOOs have not been directly addressed in the analysis, they have been shown to meet the current single failure limit, as required.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF01-03-15
Issue/Gap Description	The deterministic safety analysis for Bruce A does not distinguish between these four classes of events. The DECs introduced in REGDOC-2.5.2 are not considered in the design basis; however, the design basis includes some event sequences that would be categorized as BDBAs and meet the definition of DECs. The focus of the Safety Report is primarily on design basis events, which include design basis accidents and AOOs. The specific event classification scheme has not been applied for deterministic safety analysis and hence identified as a gap. (Gap 1)
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF01-01-15
Issue/Gap Description	Although some common-cause internally and externally initiated events form part of the design basis for the plant, these have not been explicitly addressed in the deterministic safety analysis as required in this clause. Subsequently, this is assessed as a gap. (Gap 2)
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF01-03-16
Issue/Gap Description	Although some common-cause internally and externally initiated events form part of the design basis for the plant, these have not been explicitly addressed in the deterministic safety analysis as required in this clause. Subsequently, this is assessed as a gap. (Gap 2)
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_4.2.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF01-01-16
Issue/Gap Description	The deterministic safety analysis for Bruce B does not distinguish between these four classes of events. The DECs introduced in REGDOC-2.5.2 are not considered in the design basis; however, the design basis includes some event sequences that would be categorized as BDBAs and meet the definition of DECs. The focus of the Safety Report is primarily on design basis events, which include design basis accidents and AOOs. The specific event classification scheme has not been applied for deterministic safety analysis and hence identified as a gap. (Gap 1)
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_5.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	5.3 Design control measures
Requirement Assessed	Processes, procedures and practices shall be established as part of the overall management system so as to achieve the design objectives. This shall include identifying all performance and assessment parameters for the plant design as well as detailed plans for each SSC, in order to ensure consistent quality of the design and the selected components. The design controls shall be such that the initial design, and any subsequent change or safety improvement, is carried out in accordance with established processes and procedures which call on appropriate standards and codes and address applicable requirements and design bases. Appropriate design control measures shall also facilitate identification and control of design interfaces. The adequacy of the design, including design tools and design inputs and outputs, shall be verified or validated by individuals or groups that are independent from those who originally performed the work. Verifications, validations, and approvals shall be completed before the detailed design is implemented. The computer software used for design and analysis calculations shall be qualified in accordance with applicable standards. Guidance Design control measures, in the form of processes, procedures and practices, include: • design initiation, including identification of scope • work control and planning of design activities • selection of competent staff • identification and control of design inputs • establishment of design requirements • evaluation of design concepts and selection of preferred concept • selection of design tools and computer software • conduct of conceptual safety analysis to assess preferred design concept • conduct of detailed design and production of design documentation and records • definition of any limiting conditions for safe operation • design verification and validation • configuration management • identification and control of design interfaces CSA N286, Management system requirements for nuclear power plants, is

	the Canadian standard identifying management system requirements for the design, purchasing, construction, installation, commissioning, operating, and decommissioning of NPPs. CNSC G-149, Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors, and CSA N286.7, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants, provide complementary requirements and guidance for analytical, scientific and design computer programs. Organizations from nations not using the aforementioned documents should identify the codes, standards, and specifications on which their design and safety analysis control measures are based, whether national or international – such as IAEA GS-G-3.5, The Management System for Nuclear Installations Safety Guide, referenced publications, and ISO 9001:2008 Quality Management Systems – Requirements. Such control measures should be mapped to the requisite CSA N286 clauses to demonstrate that they satisfy Canadian requirements. Where gaps are identified, the measures to address them should be described. Organizational processes and procedures can be specific to design and safety analysis, or be part of an overall management system (or quality assurance program) for other NPP lifecycle activities. In the latter case, the organization should identify those processes and procedures applicable to design and safety analysis. There are no specific platforms, styles or format requirements for documenting design control measures; however, design organizations should identify the types of documents, the style, the format and the media (paper-based, electronic or Web-based) they intend to use to control their design activities.
Macro-Gap	SF01-04-15
Issue/Gap Description	In general, the practice as defined in this clause has been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report for which validated codes have been available in the past. It is standard practice for all new safety analyses. However, the original design analyses had been produced using legacy tools predating N286.7-99. This is identified as a gap and further discussed in the clause by clause assessment against requirements of REGDOC-2.4.2 in Safety Factor 5 (Gap).
Rationale	See Notes 1 and 3

Gap #	SF01_CNSC REGDOC 2.5.2_5.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	5.3 Design control measures
Requirement Assessed	Processes, procedures and practices shall be established as part of the overall management system so as to achieve the design objectives. This shall include identifying all performance and assessment parameters for the plant design as well as detailed plans for each SSC, in order to ensure consistent quality of the design and the selected components. The design controls shall be such that the initial design, and any subsequent change or safety improvement, is carried out in accordance with established processes and procedures which call on appropriate standards and codes and address applicable requirements and design bases. Appropriate design control measures shall also facilitate identification and control of design interfaces. The adequacy of the design, including design tools and design inputs and outputs, shall be verified or validated by individuals or groups that are independent from those who originally performed the work. Verifications, validations, and approvals shall be completed before the detailed design is implemented. The computer software used for design and analysis calculations shall be qualified in accordance with applicable standards. Guidance Design control measures, in the form of processes, procedures and practices, include: • design initiation, including identification of scope • work control and planning of design activities • selection of competent staff • identification and control of design inputs • establishment of design requirements • evaluation of design concepts and selection of preferred concept • selection of design tools and computer software • conduct of conceptual safety analysis to assess preferred design concept • conduct of detailed design and production of design documentation and records • definition of any limiting conditions for safe operation • design verification and validation • configuration management • identification and control of design interfaces CSA N286, Management system requirements for nuclear power plants, is

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Macro-Gap	SF01-04-16
Issue/Gap Description	In general, the practice as defined in this clause has been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report for which validated codes have been available in the past. It is standard practice for all new safety analyses. However, the original design analyses had been produced using legacy tools predating N286.7-99. This is identified as a gap and further discussed in the clause by clause assessment against requirements of REGDOC-2.4.1 in Safety Factor 5 (Gap).
Rationale	See Notes 1 and 3

Gap #	SF01_CNSC REGDOC 2.5.2_6.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.1 Application of defence in depth
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment) and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given

	provision and the existence of supporting safety analyses. Deficiencies in the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences of the accident. The independence between all levels of defence should be achieved, in particular, through diverse provisions. The strengthening of each of these levels separately would provide, as far as reasonably achievable, an overall reinforcement of defence in depth. For example, the use of dedicated systems to deal with DECs ensures the independence of the fourth defence level.
Macro-Gap	SF01-01-15
Issue/Gap Description	The second level of defence detects and intercepts deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions. This is done by measuring deviations from normal operating conditions by both the regulating system and the special safety systems. The process features of the regulating system (liquid zone control and setback function) and the safety features (stepback function) can shut the reactor down for all but the most serious PIEs. Either of the two fully independent shutdown systems is capable of shutting the reactor down for all PIEs, should the regulating system not be able to do this. In the case of fuel overheating, the ECI system can prevent failure of the fuel sheath (barrier 2) for all but the most serious LOCAs. In regard to item (3), the ECI or moderator systems are capable of maintaining the integrity of the Heat Transport system (barrier 3) for all design basis accidents. As indicated in the compliance assessment against REGDOC-2.4.1 in Safety Factor 5, Level 2 defence in depth is not demonstrated explicitly for AOOs and is identified as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_6.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.1 Application of defence in depth
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment) and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given

	provision and the existence of supporting safety analyses. Deficiencies in the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences of the accident. The independence between all levels of defence should be achieved, in particular, through diverse provisions. The strengthening of each of these levels separately would provide, as far as reasonably achievable, an overall reinforcement of defence in depth. For example, the use of dedicated systems to deal with DECs ensures the independence of the fourth defence level.
Macro-Gap	SF01-01-16
Issue/Gap Description	As indicated in the compliance assessment against REGDOC-2.4.1 in Safety Factor 5, Level 2 defence in depth is not demonstrated explicitly for AOOs and is identified as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_6.6.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.6.1 Requirements for multiple units
Requirement Assessed	The design shall take due account of challenges to multiple units at a site. Specifically, the risk associated with common-cause events affecting more than one unit at a time shall be considered. Guidance The presence of multiple units at a site, or common-cause events could exacerbate challenges that the plant personnel would face during an accident. The events and consequences of an accident at one unit may affect the accident progression or hamper accident management activities at the neighbouring unit; available resources (personnel, equipment and consumable resources) would need to be shared among several units. These challenges should be identified and the available resources and mitigation strategies shown to be adequate.
Macro-Gap	SF01-03-15
Issue/Gap Description	A model for severe accidents with multi units is to be considered within SAMG program. Common-cause events are not analyzed explicitly in Part 3 of the Safety Report; therefore this is assessed as a gap (Gap). This gap is being prioritized to be considered early within Safety Report update towards the compliance with REGDOC-2.4.1.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_6.6.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.6.1 Requirements for multiple units
Requirement Assessed	The design shall take due account of challenges to multiple units at a site. Specifically, the risk associated with common-cause events affecting more than one unit at a time shall be considered. Guidance The presence of multiple units at a site, or common-cause events could exacerbate challenges that the plant personnel would face during an accident. The events and consequences of an accident at one unit may affect the accident progression or hamper accident management activities at the neighbouring unit; available resources (personnel, equipment and consumable resources) would need to be shared among several units. These challenges should be identified and the available resources and mitigation strategies shown to be adequate.
Macro-Gap	SF01-03-16
Issue/Gap Description	Common-cause events are not analyzed explicitly in Part 3 of the Safety Report; therefore this is assessed as a gap (Gap). This gap is being prioritized to be considered early within Safety Report update towards the compliance with REGDOC-2.4.1.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.15.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.15.3 Lifting and handling of large loads
Requirement Assessed	The lifting and handling of large and heavy loads, particularly those containing radioactive material, shall be considered in the NPP design. This shall include identification of the large loads, traversing routes and situations where they need to be lifted over areas of the plant that are critical to safety. The design of all cranes and lifting devices shall, therefore, incorporate large margins, appropriate interlocks, and other safety features to accommodate the lifting of large loads. The drop of large loads lifted and handled in areas where there are systems and components that are important to safety shall be taken into account in the design. The potential load due to the large load drop shall be taken into account in the analysis of DBAs.
Macro-Gap	SF01-10-15
Issue/Gap Description	The Bruce A design does not consider the drop of large loads in areas where systems and components important to safety are located. There is no documented corresponding analysis to justify safe operation. Therefore, it is assessed as a gap. (Gap 2).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.15.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.15.3 Lifting and handling of large loads
Requirement Assessed	The lifting and handling of large and heavy loads, particularly those containing radioactive material, shall be considered in the NPP design. This shall include identification of the large loads, traversing routes and situations where they need to be lifted over areas of the plant that are critical to safety. The design of all cranes and lifting devices shall, therefore, incorporate large margins, appropriate interlocks, and other safety features to accommodate the lifting of large loads. The drop of large loads lifted and handled in areas where there are systems and components that are important to safety shall be taken into account in the design. The potential load due to the large load drop shall be taken into account in the analysis of DBAs.
Macro-Gap	SF01-10-16
Issue/Gap Description	The Bruce B design does not consider the drop of large loads in areas where systems and components important to safety are located. There is no documented corresponding analysis to justify safe operation when such loads are present. Therefore, it is assessed as a gap. (Gap 2).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.3.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.3.4 Design extension conditions
Requirement Assessed	The design authority shall identify the set of design-extension conditions (DECs) based on deterministic and probabilistic methods, operational experience, engineering judgment and the results of research and analysis. These DECs shall be used to further improve the safety of the NPP by enhancing the plant's capabilities to withstand, without significant radiological releases, accidents that are either more severe than DBAs or that involve additional failures. The design shall be such that plant states that could lead to significant radioactive releases are practically eliminated. For plant states that are not practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures. Complementary design features shall be provided to cope with DECs. Their design shall be based on a combination of phenomenological models, engineering judgments, and probabilistic methods. The rules and practices that have been applied to the complementary design features shall be identified. These rules and practices do not necessarily need to incorporate the same degree of conservatism as those applied to the design basis. The design shall identify a radiological and combustible gas accident source term, for use in the specification of the complementary design features for DECs. This source term is referred to as the reference source term and shall be based on a set of representative core damage accidents established by the design authority. To the extent practicable, the design shall provide biological shielding of appropriate composition and thickness in order to protect operational personnel during DECs. In the case of plants with multiple units at a site, the use of available support from other units shall only be relied upon if the safe operation of the other units is not compromised. Guidance DECs are the subset of BDBAs that are considered in the design. BDBAs are all events less frequent than DBAs; there is no lower frequency bound. For identifying DECs, consideration should be given to:

	• factors of the accident progression (i.e., physical conditions, processes and phenomena) • BDBA (including severe accident) scenarios resulting from initiating events, human actions, and SSC operability (success or failure) • selection of bounding events that are considered in design and determination of limiting values and ranges of the parameters of these events The design should identify the features that are designed for use in, or that are capable of preventing or mitigating events considered in DECs. These features include complementary design features and other SSCs that may be credited for DECs. These features should: 1. be independent, to the extent practicable, of those used in more frequent accidents 2. have a reliability commensurate with the function that they are required to fulfill The choice of the DECs to be analyzed should be explained and justified, indicating whether it has been made on the basis of a PSA or other analysis that identifies potential vulnerabilities of the plant. For use in the specification of the complementary design features for DECs, the reference source term should be calculated for a set of representative accident scenarios based on the best-estimate models. This should take into account the uncertainties of key parameters and the possible changes in governing physical processes. Accidents in this category are, typically, sequences involving more than one failure (unless these are taken into account in the DBAs at the design stage). Such sequences may include DBAs with degraded performance of a safety system, and sequences that could lead to containment bypass. The analysis of those accidents may: • use best-estimate models and assumptions • take credit for realistic system action and performance beyond original intended functions, including the potential use of safety, non-safety and temporary systems • take credit for realistic operator actions Where this is not possible, reasonably conservative assumptions should be made in which the uncertainties in the understanding of the physical processes being modelled are considered. The analysis should justify the approach taken. Accident conditions with a significant release are considered to have been practically eliminated: • if it is physically impossible for the condition to occur, or

	• if the condition can be considered with a high degree of confidence to be extremely unlikely to arise Physical impossibility can be demonstrated by a design feature that would preclude initiation or further progress of an accident scenario. Care should be taken when assumptions are used to support the demonstration. Such assumptions should be adequately acknowledged and addressed. To demonstrate practical elimination as extremely unlikely with a high degree of confidence, the following should be considered: • The degree of substantiation provided for the demonstration of practical elimination should take account of the assessed frequency of the situation to be eliminated and of the degree of confidence in the assessed frequency. • Practical elimination of an accident should not be claimed solely based on compliance with a probabilistic cut-off value. Even if the probability of an accident sequence is very low, any additional design features, operational measures or accident management procedures to lower the risk further should be implemented to the extent practicable. • The most stringent requirements regarding the demonstration of practical elimination should apply in the case of an event with the potential to lead directly to a severe accident; i.e., from Level 1 to Level 4 for defence in depth. For example, demonstration of practical elimination of a heterogeneous boron dilution event in a pressurized water reactor (PWR) would require a detailed substantiation. • The necessary high confidence in low likelihood should, wherever possible, be supported by means such as: • multiple layers of protection • application of the safety principles of independence, diversity, separation, redundancy • use of passive safety features • use of multiple independent controls • It should be ensured that the practical elimination provisions remain in place and valid throughout the plant lifetime; for example, through in-service and periodic inspections. In each case, the demonstration should show sufficient knowledge of the accident sequence analyzed and of the phenomena involved, substantiated by relevant evidence. To minimize uncertainties and to increase the robustness of a plant's safety case, demonstration of practical elimination should preferably rely on the criterion of physical impossibility, rather than the second probabilistic criterion (extreme unlikelihood with high confidence). Portable equipment should be classified based on its safety importance. There may be different options available to fulfill the fundamental safety

	functions during DECs. However, when called upon the portable onsite or offsite equipment credited is expected to be effective with reasonable confidence. Portable onsite or offsite equipment may be one of the means for mitigation in support of the severe accident management guidelines.
Macro-Gap	SF01-01-15
Issue/Gap Description	A gap is identified in Section 4.2.3 of Compliance assessment against REGDOC-2.4.1. The current deterministic safety analysis as documented in Part 3 of the Safety Report does not distinguish between these three classes of events. The focus of the Safety Report is primarily on design basis events, which include design basis accidents and AOOs. The specific event classification scheme has not been applied for deterministic safety analysis (Gap). Further details are presented in Safety Factor 5.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.3.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.3.4 Design extension conditions
Requirement Assessed	The design authority shall identify the set of design-extension conditions (DECs) based on deterministic and probabilistic methods, operational experience, engineering judgment and the results of research and analysis. These DECs shall be used to further improve the safety of the NPP by enhancing the plant's capabilities to withstand, without significant radiological releases, accidents that are either more severe than DBAs or that involve additional failures. The design shall be such that plant states that could lead to significant radioactive releases are practically eliminated. For plant states that are not practically eliminated, only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement these measures. Complementary design features shall be provided to cope with DECs. Their design shall be based on a combination of phenomenological models, engineering judgments, and probabilistic methods. The rules and practices that have been applied to the complementary design features shall be identified. These rules and practices do not necessarily need to incorporate the same degree of conservatism as those applied to the design basis. The design shall identify a radiological and combustible gas accident source term, for use in the specification of the complementary design features for DECs. This source term is referred to as the reference source term and shall be based on a set of representative core damage accidents established by the design authority. To the extent practicable, the design shall provide biological shielding of appropriate composition and thickness in order to protect operational personnel during DECs. In the case of plants with multiple units at a site, the use of available support from other units shall only be relied upon if the safe operation of the other units is not compromised. Guidance DECs are the subset of BDBAs that are considered in the design. BDBAs are all events less frequent than DBAs; there is no lower frequency bound. For identifying DECs, consideration should be given to:

	• factors of the accident progression (i.e., physical conditions, processes and phenomena) • BDBA (including severe accident) scenarios resulting from initiating events, human actions, and SSC operability (success or failure) • selection of bounding events that are considered in design and determination of limiting values and ranges of the parameters of these events The design should identify the features that are designed for use in, or that are capable of preventing or mitigating events considered in DECs. These features include complementary design features and other SSCs that may be credited for DECs. These features should: 1. be independent, to the extent practicable, of those used in more frequent accidents 2. have a reliability commensurate with the function that they are required to fulfill The choice of the DECs to be analyzed should be explained and justified, indicating whether it has been made on the basis of a PSA or other analysis that identifies potential vulnerabilities of the plant. For use in the specification of the complementary design features for DECs, the reference source term should be calculated for a set of representative accident scenarios based on the best-estimate models. This should take into account the uncertainties of key parameters and the possible changes in governing physical processes. Accidents in this category are, typically, sequences involving more than one failure (unless these are taken into account in the DBAs at the design stage). Such sequences may include DBAs with degraded performance of a safety system, and sequences that could lead to containment bypass. The analysis of those accidents may: • use best-estimate models and assumptions • take credit for realistic system action and performance beyond original intended functions, including the potential use of safety, non-safety and temporary systems • take credit for realistic operator actions Where this is not possible, reasonably conservative assumptions should be made in which the uncertainties in the understanding of the physical processes being modelled are considered. The analysis should justify the approach taken. Accident conditions with a significant release are considered to have been practically eliminated: • if it is physically impossible for the condition to occur, or

	• if the condition can be considered with a high degree of confidence to be extremely unlikely to arise Physical impossibility can be demonstrated by a design feature that would preclude initiation or further progress of an accident scenario. Care should be taken when assumptions are used to support the demonstration. Such assumptions should be adequately acknowledged and addressed. To demonstrate practical elimination as extremely unlikely with a high degree of confidence, the following should be considered: • The degree of substantiation provided for the demonstration of practical elimination should take account of the assessed frequency of the situation to be eliminated and of the degree of confidence in the assessed frequency. • Practical elimination of an accident should not be claimed solely based on compliance with a probabilistic cut-off value. Even if the probability of an accident sequence is very low, any additional design features, operational measures or accident management procedures to lower the risk further should be implemented to the extent practicable. • The most stringent requirements regarding the demonstration of practical elimination should apply in the case of an event with the potential to lead directly to a severe accident; i.e., from Level 1 to Level 4 for defence in depth. For example, demonstration of practical elimination of a heterogeneous boron dilution event in a pressurized water reactor (PWR) would require a detailed substantiation. • The necessary high confidence in low likelihood should, wherever possible, be supported by means such as: • multiple layers of protection • application of the safety principles of independence, diversity, separation, redundancy • use of passive safety features • use of multiple independent controls • It should be ensured that the practical elimination provisions remain in place and valid throughout the plant lifetime; for example, through in-service and periodic inspections. In each case, the demonstration should show sufficient knowledge of the accident sequence analyzed and of the phenomena involved, substantiated by relevant evidence. To minimize uncertainties and to increase the robustness of a plant's safety case, demonstration of practical elimination should preferably rely on the criterion of physical impossibility, rather than the second probabilistic criterion (extreme unlikelihood with high confidence). Portable equipment should be classified based on its safety importance. There may be different options available to fulfill the fundamental safety

	functions during DECs. However, when called upon the portable onsite or offsite equipment credited is expected to be effective with reasonable confidence. Portable onsite or offsite equipment may be one of the means for mitigation in support of the severe accident management guidelines.
Macro-Gap	SF01-01-16
Issue/Gap Description	A gap is identified in Section 4.2.3 of Compliance assessment against REGDOC-2.4.1. The current deterministic safety analysis as documented in Part 3 of the Safety Report does not distinguish between these three classes of events. The focus of the Safety Report is primarily on design basis events, which include design basis accidents and AOOs. The specific event classification scheme has not been applied for deterministic safety analysis (Gap). Further details are presented in Safety Factor 5.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.3 Plant states
Requirement Assessed	Plant states considered in the design shall be grouped into the following four categories: 1. Normal operation is an operation within specified OLCs, including start-up, power operation, shutting down, shutdown, maintenance, testing, and refuelling. 2. An anticipated operational occurrence (AOO) is a deviation from normal operation that is expected to occur once or several times during the operating lifetime of the NPP but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety, or lead to accident conditions. 3. Design-basis accidents (DBAs) are accident conditions for which an NPP is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within regulated limits. 4. Design extension conditions (DECs) are a subset of beyond-design-basis accidents that are considered in the design process of the facility in accordance with best-estimate methodology to keep releases of radioactive material within acceptable limits. Design extension conditions could include severe accidents. Acceptance criteria shall be assigned to each plant state considered in the design, taking into account the principle that frequent PIEs will have only minor or no radiological consequences, and that any events that may result in severe consequences will be of extremely low probability. Guidance Plant states considered in the design are divided into normal operation, AOOs, DBAs and DECs. The design requirements of SSCs should then be developed to ensure that the plant is capable of meeting applicable deterministic and probabilistic requirements for each plant state. Note that the plant states diagram in section 7.2 identifies BDBA as a plant state. However, only a subset of BDBAs is considered in the design. These are DECs. The design should include the following: • criteria for transition to normal operation following an AOO or DBA (e.g., the safety functions are provided, and the OLC limits for the operating configurations are met)

	• key parameters and characteristics for operational states, including nominal values and deviations due to uncertainties and settings of instruments, controls, trips, equipment action time, or due to process fluctuations • permissible conditions for different operating configurations (e.g., cold and pressurized) including transient time (e.g., power level of reactor or turbine, normal planned power transient rate, heat-up and cool-down rates) for the NPP's operating life • methods of transferring the plant between different operating configurations • final safe configurations after AOOs, DBAs, and DECs
Macro-Gap	SF01-01-15
Issue/Gap Description	As noted before the AOOs are not explicitly covered in the existing design documentation; therefore this is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.3 Plant states
Requirement Assessed	Plant states considered in the design shall be grouped into the following four categories: 1. Normal operation is an operation within specified OLCs, including start-up, power operation, shutting down, shutdown, maintenance, testing, and refuelling. 2. An anticipated operational occurrence (AOO) is a deviation from normal operation that is expected to occur once or several times during the operating lifetime of the NPP but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety, or lead to accident conditions. 3. Design-basis accidents (DBAs) are accident conditions for which an NPP is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within regulated limits. 4. Design extension conditions (DECs) are a subset of beyond-design-basis accidents that are considered in the design process of the facility in accordance with best-estimate methodology to keep releases of radioactive material within acceptable limits. Design extension conditions could include severe accidents. Acceptance criteria shall be assigned to each plant state considered in the design, taking into account the principle that frequent PIEs will have only minor or no radiological consequences, and that any events that may result in severe consequences will be of extremely low probability. Guidance Plant states considered in the design are divided into normal operation, AOOs, DBAs and DECs. The design requirements of SSCs should then be developed to ensure that the plant is capable of meeting applicable deterministic and probabilistic requirements for each plant state. Note that the plant states diagram in section 7.2 identifies BDBA as a plant state. However, only a subset of BDBAs is considered in the design. These are DECs. The design should include the following: • criteria for transition to normal operation following an AOO or DBA (e.g., the safety functions are provided, and the OLC limits for the operating configurations are met)

	• key parameters and characteristics for operational states, including nominal values and deviations due to uncertainties and settings of instruments, controls, trips, equipment action time, or due to process fluctuations • permissible conditions for different operating configurations (e.g., cold and pressurized) including transient time (e.g., power level of reactor or turbine, normal planned power transient rate, heat-up and cool-down rates) for the NPP's operating life • methods of transferring the plant between different operating configurations • final safe configurations after AOOs, DBAs, and DECs
Macro-Gap	SF01-01-16
Issue/Gap Description	As noted before the AOOs are not explicitly covered in the existing design documentation; therefore this is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4 Postulated initiating events
Requirement Assessed	The design for the NPP shall apply a systematic approach to identifying a comprehensive set of postulated initiating events, such that all foreseeable events with the potential for serious consequences or with a significant frequency of occurrence are anticipated and considered. Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs as well as operator errors, common-cause internal hazards, and external hazards. For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site. Guidance The postulated initiating events (PIEs) are identified using engineering judgment and deterministic and probabilistic assessment. A justification of the extent of usage of deterministic safety analyses and probabilistic safety analyses should be provided, in order to show that all foreseeable events have been considered. Sufficient information should be provided regarding the methods used to identify PIEs, their scope and classification. In cases where the identification methods have made use of analytical tools (e.g., master logic diagrams, hazard and operability analysis, failure modes and effect analysis), detailed information is expected to be presented. A systematic approach to event classification should consider all internal and external events, all normal operating configurations, various plant and site conditions, and failure in other plant systems (e.g., storage for irradiated fuel, and tanks for radioactive substances). The design should take into account failure of equipment that is not part of the NPP, if the failure has a significant impact on nuclear safety. CNSC REGDOC-2.4.1, Deterministic Safety Analysis and REGDOC-2.4.2, Probabilistic Safety Assessments, provide the requirements and guidance for establishing the scope of PIEs, and for classifying the PIEs in accordance with their anticipated frequencies, and other factors, as appropriate. For further information on the safety analysis for the identified PIEs, refer to section 9.0 of this document.
Macro-Gap	SF01-01-15

Issue/Gap Description	A systematic event identification process is not well documented and/or demonstrated; therefore this is assessed as a gap (Gap). Postulated initiating events are not categorized into AOOs, DBAs or BDBAs. Additional details are provided in the assessment against REGDOC-2.4.1.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4 Postulated initiating events
Requirement Assessed	The design for the NPP shall apply a systematic approach to identifying a comprehensive set of postulated initiating events, such that all foreseeable events with the potential for serious consequences or with a significant frequency of occurrence are anticipated and considered. Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs as well as operator errors, common-cause internal hazards, and external hazards. For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site. Guidance The postulated initiating events (PIEs) are identified using engineering judgment and deterministic and probabilistic assessment. A justification of the extent of usage of deterministic safety analyses and probabilistic safety analyses should be provided, in order to show that all foreseeable events have been considered. Sufficient information should be provided regarding the methods used to identify PIEs, their scope and classification. In cases where the identification methods have made use of analytical tools (e.g., master logic diagrams, hazard and operability analysis, failure modes and effect analysis), detailed information is expected to be presented. A systematic approach to event classification should consider all internal and external events, all normal operating configurations, various plant and site conditions, and failure in other plant systems (e.g., storage for irradiated fuel, and tanks for radioactive substances). The design should take into account failure of equipment that is not part of the NPP, if the failure has a significant impact on nuclear safety. CNSC REGDOC-2.4.1, Deterministic Safety Analysis and REGDOC-2.4.2, Probabilistic Safety Assessments, provide the requirements and guidance for establishing the scope of PIEs, and for classifying the PIEs in accordance with their anticipated frequencies, and other factors, as appropriate. For further information on the safety analysis for the identified PIEs, refer to section 9.0 of this document.
Macro-Gap	SF01-01-16

Issue/Gap Description	A systematic event identification process is not well documented and/or demonstrated; therefore this is assessed as a gap (Gap). Postulated initiating events are not categorized into AOOs, DBAs or BDBAs. Additional details are provided in the assessment against REGDOC-2.4.1 documented in Safety Factor 5.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.7_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.7 Pressure-retaining structures, systems and components
Requirement Assessed	All pressure-retaining SSCs shall be protected against overpressure conditions, and shall be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. For DECs, relief capacity shall be sufficient to provide reasonable confidence that pressure boundaries credited in severe accident management will not fail. All pressure-retaining SSCs of the reactor coolant system and auxiliaries shall be designed with an appropriate safety margin to ensure that the pressure boundary will not be breached, and that fuel design limits will not be exceeded in operational states, or DBA conditions. The design shall minimize the likelihood of flaws in pressure boundaries. This shall include timely detection of flaws in pressure boundaries important to safety. Unless otherwise justified, all pressure boundary SSCs shall be designed to withstand static and dynamic loads anticipated in operational states, and DBAs. SSC design shall include protection against postulated pipe ruptures, unless otherwise justified. The operation of pressure relief devices shall not lead to significant radioactive releases from the plant. Where two fluid systems operating at different pressures are interconnected, failure of the interconnection shall be considered. Both systems shall either be designed to withstand the higher pressure, or provision shall be made so that the design pressure of the system operating at the lower pressure will not be exceeded. Adequate isolation shall be provided at the interfaces between the reactor coolant system and connecting systems operating at lower pressures, in order to prevent the overpressure of such systems and possible loss-of-coolant accidents. Consideration shall be given to the characteristics and importance of the isolation and its reliability targets. Isolation devices shall be either closed or close automatically on demand. The response time and speed of closure shall be in accordance with the acceptance criteria defined for postulated initiating events. All pressure boundary piping and vessels shall be separated from electrical and control systems to the greatest extent practicable. Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries

	throughout the design life. If full inspection is not achievable, then it shall be augmented by indirect methods such as a program of surveillance of reference components. Leak detection is an acceptable method when the SSC is leak-before-break qualified. Guidance For the design of pressure-retaining systems and components, the design authority should ensure the selection of codes and standards is commensurate with the safety class and is adequate to provide confidence that plant failures are minimized. This is achieved by using industry standards - such as CSA N285, General requirements for pressure-retaining systems and components in CANDU nuclear power plants and ASME Boiler and Pressure Vessel Code - to meet the requirements of different classes of pressure-retaining systems, components, piping and their supports. Alternative codes and standards may be used if this would result in an equivalent or superior level of safety; justifications should be provided in such cases. The design should make provisions to limit stresses and deformation of SSCs important to safety during and after PIEs. The list of PIEs should be comprehensive, and the loads generated by them should be included in the design analysis. The loads generated by these PIEs should be included in the stress analyses required by the design. REGDOC-2.5.2 requires the design to minimize the likelihood of flaws in pressure boundaries. For example, the reactor coolant pressure boundary should be designed with sufficient margin to ensure that, under all operating configurations, the material selected will behave in a non-brittle manner and minimize the probability of rapidly propagating fractures. The pressure boundary components in an NPP almost invariably contain process fluids at very high temperature and pressure. The design should take into account the location of high-energy lines in relation to SSCs important to safety, in order to limit or reduce pipe whip concerns. This includes consideration, where applicable, of items such as: o components in the means of shutdown o main coolant pumps o headers o emergency core cooling system components o steam generators o steam lines o turbine Leak-before-break A qualified leak-before-break (LBB) system design will permit the design authority to optimize protective hardware - such as pipe whip restraints

and jet impingement barriers - and to redesign pipe-connected components, their supports and their internals.

A qualified LBB methodology should include the following:

o LBB should be only applied to high-energy, ASME Code Class 1 or 2 piping or the equivalent. Applications to other high-energy piping may be performed based on an evaluation of the proposed design and in-service inspection requirements.
O No uncontrolled active degradation mechanism should exist in the piping system to be qualified for LBB.
O An evaluation of phenomena such as water hammer, creep damage, flow accelerated
corrosion and fatigue should be performed to cover the entire life of the high-energy piping systems. To demonstrate that water hammer is not a significant contributor to pipe rupture, reliance on historical frequencies of water hammer events in specific piping systems coupled with reviews of operating procedures and conditions may be used for this evaluation.
O Leak detection methods for the reactor coolant should ensure that adequate detection margins exist for the postulated through-wall flaw used in the deterministic fracture mechanics evaluation. The margins should cover uncertainties in the determination of leakage from a piping system.
O Stress analyses of the piping that is considered for LBB should be in accordance with the requirements of section III of the ASME code or equivalent.
O The LBB evaluation should use design basis loads and, after construction, be updated to use the as-built piping configuration, as opposed to the design configuration.
O The methodology should take account of potential for degradation by erosion, corrosion, and erosion-cavitation due to unfavourable flow conditions and water chemistry.
O The methodology should take account of material susceptibility to corrosion, the potential for high residual stresses, and environmental conditions that could lead to degradation by stress
corrosion cracking.

In addition, leak detection methods for the reactor coolant should be examined so as to ensure that adequate detection margins exist for the postulated through-wall flaw used in the deterministic fracture mechanics evaluation.

Finite element methods

The design authority customarily uses finite element methods to show that all of the pressure boundary components (both vessels and piping) meet the structural integrity requirements imposed by applicable design codes and standards. When finite element methods are used for design analyses covering all ASME (or equivalent) class components, the design authority should ensure that:

	o finite element modelling and analysis assumptions are checked to make sure they are justified and conservative o finite element mesh is properly refined to account for geometric structural discontinuities with proper element shapes and aspect ratios o loads and boundary conditions are correct and properly applied in the finite element models o load combinations and scale factors applied to unit load cases conform to design or load specifications o linearized stress results, obtained from load combinations, are compared with ASME code (or equivalent) allowable limits
Macro-Gap	SF01-06-15
Issue/Gap Description	A review of the same clause in RD-337 indicated that the Bruce A design does not fully meet this requirement, as documented in [NK21-CORR-00531-11005]. The Safety Report for Bruce A (NK21-SR-01320-00003, Rev. 004) includes a wide range of accidents that are considered to be AOOs, although no credit is taken for control system protective action. Since there is not a systematic analysis of the control system capability to cope with AOOs, no definitive statement can be made in regard to the compliance with the AOO section of this clause (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_7.7_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.7 Pressure-retaining structures, systems and components
Requirement Assessed	All pressure-retaining SSCs shall be protected against overpressure conditions, and shall be classified, designed, fabricated, erected, inspected, and tested in accordance with established standards. For DECs, relief capacity shall be sufficient to provide reasonable confidence that pressure boundaries credited in severe accident management will not fail. All pressure-retaining SSCs of the reactor coolant system and auxiliaries shall be designed with an appropriate safety margin to ensure that the pressure boundary will not be breached, and that fuel design limits will not be exceeded in operational states, or DBA conditions. The design shall minimize the likelihood of flaws in pressure boundaries. This shall include timely detection of flaws in pressure boundaries important to safety. Unless otherwise justified, all pressure boundary SSCs shall be designed to withstand static and dynamic loads anticipated in operational states, and DBAs. SSC design shall include protection against postulated pipe ruptures, unless otherwise justified. The operation of pressure relief devices shall not lead to significant radioactive releases from the plant. Where two fluid systems operating at different pressures are interconnected, failure of the interconnection shall be considered. Both systems shall either be designed to withstand the higher pressure, or provision shall be made so that the design pressure of the system operating at the lower pressure will not be exceeded. Adequate isolation shall be provided at the interfaces between the reactor coolant system and connecting systems operating at lower pressures, in order to prevent the overpressure of such systems and possible loss-of-coolant accidents. Consideration shall be given to the characteristics and importance of the isolation and its reliability targets. Isolation devices shall be either closed or close automatically on demand. The response time and speed of closure shall be in accordance with the acceptance criteria defined for postulated initiating events. All pressure boundary piping and vessels shall be separated from electrical and control systems to the greatest extent practicable. Pressure-retaining components whose failure will affect nuclear safety shall be designed to permit inspection of their pressure boundaries

	throughout the design life. If full inspection is not achievable, then it shall be augmented by indirect methods such as a program of surveillance of reference components. Leak detection is an acceptable method when the SSC is leak-before-break qualified. Guidance For the design of pressure-retaining systems and components, the design authority should ensure the selection of codes and standards is commensurate with the safety class and is adequate to provide confidence that plant failures are minimized. This is achieved by using industry standards - such as CSA N285, General requirements for pressure-retaining systems and components in CANDU nuclear power plants and ASME Boiler and Pressure Vessel Code - to meet the requirements of different classes of pressure-retaining systems, components, piping and their supports. Alternative codes and standards may be used if this would result in an equivalent or superior level of safety; justifications should be provided in such cases. The design should make provisions to limit stresses and deformation of SSCs important to safety during and after PIEs. The list of PIEs should be comprehensive, and the loads generated by them should be included in the design analysis. The loads generated by these PIEs should be included in the stress analyses required by the design. REGDOC-2.5.2 requires the design to minimize the likelihood of flaws in pressure boundaries. For example, the reactor coolant pressure boundary should be designed with sufficient margin to ensure that, under all operating configurations, the material selected will behave in a non-brittle manner and minimize the probability of rapidly propagating fractures. The pressure boundary components in an NPP almost invariably contain process fluids at very high temperature and pressure. The design should take into account the location of high-energy lines in relation to SSCs important to safety, in order to limit or reduce pipe whip concerns. This includes consideration, where applicable, of items such as: o components in the means of shutdown o main coolant pumps o headers o emergency core cooling system components o steam generators o steam lines o turbine Leak-before-break A qualified leak-before-break (LBB) system design will permit the design authority to optimize protective hardware - such as pipe whip restraints

	and jet impingement barriers - and to redesign pipe-connected components, their supports and their internals. A qualified LBB methodology should include the following: o LBB should be only applied to high-energy, ASME Code Class 1 or 2 piping or the equivalent. Applications to other high-energy piping may be performed based on an evaluation of the proposed design and in-service inspection requirements. O No uncontrolled active degradation mechanism should exist in the piping system to be qualified for LBB. O An evaluation of phenomena such as water hammer, creep damage, flow accelerated corrosion and fatigue should be performed to cover the entire life of the high-energy piping systems. To demonstrate that water hammer is not a significant contributor to pipe rupture, reliance on historical frequencies of water hammer events in specific piping systems coupled with reviews of operating procedures and conditions may be used for this evaluation. O Leak detection methods for the reactor coolant should ensure that adequate detection margins exist for the postulated through-wall flaw used in the deterministic fracture mechanics evaluation. The margins should cover uncertainties in the determination of leakage from a piping system. O Stress analyses of the piping that is considered for LBB should be in accordance with the requirements of section III of the ASME code or equivalent. O The LBB evaluation should use design basis loads and, after construction, be updated to use the as-built piping configuration, as opposed to the design configuration. O The methodology should take account of potential for degradation by erosion, corrosion, and erosion-cavitation due to unfavourable flow conditions and water chemistry. O The methodology should take account of material susceptibility to corrosion, the potential for high residual stresses, and environmental conditions that could lead to degradation by stress corrosion cracking. In addition, leak detection methods for the reactor coolant should be examined so as to ensure that adequate detection margins exist for the postulated through-wall flaw used in the deterministic fracture mechanics evaluation. Finite element methods The design authority customarily uses finite element methods to show that all of the pressure boundary components (both vessels and piping) meet the structural integrity requirements imposed by applicable design codes and standards. When finite element methods are used for design analyses covering all ASME (or equivalent) class components, the design authority should ensure that:

	o finite element modelling and analysis assumptions are checked to make sure they are justified and conservative o finite element mesh is properly refined to account for geometric structural discontinuities with proper element shapes and aspect ratios o loads and boundary conditions are correct and properly applied in the finite element models o load combinations and scale factors applied to unit load cases conform to design or load specifications o linearized stress results, obtained from load combinations, are compared with ASME code (or equivalent) allowable limits
Macro-Gap	SF01-06-16
Issue/Gap Description	The Safety Report for Bruce B includes a wide range of accidents that are considered to be AOOs, although no credit is taken for control system protective action. Since there is not a systematic analysis of the control system capability to cope with AOOs, no definitive statement can be made in regard to the compliance with the AOO section of this clause (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_8.1.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.1.1 Fuel elements, assemblies and design
Requirement Assessed	Fuel assembly design shall include all components in the assembly, such as the fuel matrix, cladding, spacers, support plates, movable rods inside the assembly etc. The fuel assembly design shall also identify all interfacing systems. Fuel assemblies and the associated components shall be designed to withstand the anticipated irradiation and environmental conditions in the reactor core, and all processes of deterioration that can occur in operational states. The fuel shall remain suitable for continued use after AOOs. At the design stage, consideration shall be given to long-term storage of irradiated fuel assemblies after discharge from the reactor. Fuel design limits shall be established to include, as a minimum, limits on fuel power or temperature, limits on fuel burnup, and limits on the leakage of fission products in the reactor cooling system. The design limits shall reflect the importance of preserving the fuel matrix and cladding, as these are first and second barriers to fission product release, respectively. The design shall account for all known degradation mechanisms, with allowance being made for uncertainties in data, calculations, and fuel fabrication. Fuel assemblies shall be designed to permit adequate inspection of their structures and components prior to and following irradiation. In DBAs, the fuel assembly and its component parts shall remain in position with no distortion that would prevent effective post-accident core cooling or interfere with the actions of reactivity control devices or mechanisms. The design shall specify the acceptance criteria necessary to meet these requirements in DBAs. The requirements for reactor and fuel assembly design shall apply in the event of changes in fuel management strategy, or in operating conditions, over the lifetime of the plant. Fuel design and design limits shall reflect a verified and auditable knowledge base. The fuel shall be qualified for operation, either through experience with the same type of fuel in other reactors, or through a program of experimental testing and analysis, to ensure that fuel assembly requirements are met. Guidance The fuel design and qualification should provide assurance that the reactor

	core design requirements in section 8.1 are met. Acceptance criteria should be established for fuel damage, fuel rod failure, and fuel coolability. These criteria should be derived from experiments that identify the limitations of the material properties of the fuel and fuel assembly, and related analyses. The fuel design criteria and other design considerations are discussed below. Fuel damage Fuel damage criteria should be established for all known damage mechanisms in operational states (normal operation and AOOs). The damage criteria should assure that fuel dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. The criteria should include stress, strain or loading limits, the cumulative number of strain fatigue cycles, fretting wear, oxidation, hydriding (deuteriding in CANDU reactors), build-up of corrosion products, dimensional changes, rod internal gas pressures, worst-case hydraulic loads, and LWR control rod insertability. Fuel rod failure Fuel rod failure applies to operational states, DBAs and DECs. Fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. The design should ensure that fuel does not fail as a result of specific causes during operational states. Fuel rod failures could occur during DBAs and DECs, and are accounted for in the safety analysis. Assessment methods should be stated for, fuel failure mechanisms, reactor loading and power manoeuvring limitations, and fuel duty which lead to an acceptably low probability of failure. When applicable, the fuel rod failure criteria should consider high burnup effects, based on data of irradiated material properties. The criteria should include: • hydriding • cladding collapse • cladding overheating • fuel pellet overheating • excessive fuel enthalpy • pellet-clad interaction • stress-corrosion cracking • cladding bursting • mechanical fracturing Fuel coolability Fuel coolability applies to DBAs and, to the extent practicable, DECs. Fuel coolability criteria should be provided for all damage mechanisms in DBAs

	and DECs. The fuel should be designed to ensure that fuel rod damage will not interfere with effective emergency core cooling. The cladding temperatures should not reach a temperature high enough to allow a significant metal- water reaction to occur, thereby minimizing the potential for fission product release. The criteria should include cladding embrittlement, fuel rod ballooning, structural deformation and, in CANDU, beryllium braze penetration. Other considerations The design should also include: • all expected fuel handling activities • the effects of post-irradiation fuel assembly handling • cooling flow of other components of LWR fuel assembly (such as control rods, poison rods, instrumentation, or neutron sources) Testing, inspection, and surveillance programs Programs for testing and inspection of new fuel as well as for online fuel monitoring and post- irradiation surveillance of irradiated fuel should be established. Fuel specification The design should establish the specification of fuel rods and assembly (including LWR control rods) in order to minimize design deviations and to determine whether all design bases are met (such as limits and tolerances). Reactor core thermal hydraulic design The thermalhydraulic design should be such that sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the reactor coolant system, to prevent fuel sheath overheating. The design requirements can be demonstrated by meeting a set of derived acceptance criteria, as required by REGDOC-2.4.1, Deterministic Safety Analysis. Critical heat flux (CHF) is defined as the heat flux at departure from nucleate boiling (DNB), commonly used in pressurized water reactors (PWRs), or at dryout, commonly used in CANDU designs. It should be noted that, although a thermal margin criterion is sufficient to demonstrate that overheating from a deficient cooling mechanism can be avoided; other mechanistic methods may be acceptable as CHF is not considered as a failure mechanism. In some designs, CHF conditions during transients can be tolerated if it can be shown by other methods that the sheath temperatures do not exceed well-defined acceptable limits. However, any other criteria than the CHF criterion should address sheath

	temperature, pressure, time duration, oxidation, embrittlement etc., and these new criteria should be supported by sufficient experimental and analytical evidence. In the absence of such evidence, the core thermal-hydraulic design is expected to demonstrate a thermal margin to CHF. The demonstration of thermal margin is expected to be presented in a manner that accounts for all possible reactor operational states and conditions, as determined from operating maps including all AOOs. The demonstration should also include long term effects of plant aging and other expected changes to core configuration over the operating life of the plant. The demonstration of thermal margin should thoroughly address uncertainties of various parameters affecting the thermal margin. The design should identify all sources of significant uncertainties that contribute to the uncertainty of thermal margin. The uncertainty for each of the sources should be quantified with supportable evidence. In addition to the demonstration of thermal margin, the core thermal-hydraulic design should also address possible core power and flow oscillations and thermal-hydraulic instabilities. The design should be such that power and flow oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.
Macro-Gap	SF01-01-15
Issue/Gap Description	Qualitative acceptance criteria have been established to assess fuel and fuel channel integrity fitness-for-service (FFS) following an AOO. The AOO Fuel and Pressure Tube Fitness-For-Service Criteria for LOF, SLOCA and Slow LORC [COG-12-2049/CG402-RP-001 R01] document assesses fuel and fuel channel behaviour during an AOO event. Demonstration that fuel will remain fit for service after AOO cannot be confirmed in the current design documentation. Acceptance criteria and corresponding assessments, including inspection requirements, return to service requirements or further assessments are not available; therefore this is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_8.1.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.1.1 Fuel elements, assemblies and design
Requirement Assessed	Fuel assembly design shall include all components in the assembly, such as the fuel matrix, cladding, spacers, support plates, movable rods inside the assembly etc. The fuel assembly design shall also identify all interfacing systems. Fuel assemblies and the associated components shall be designed to withstand the anticipated irradiation and environmental conditions in the reactor core, and all processes of deterioration that can occur in operational states. The fuel shall remain suitable for continued use after AOOs. At the design stage, consideration shall be given to long-term storage of irradiated fuel assemblies after discharge from the reactor. Fuel design limits shall be established to include, as a minimum, limits on fuel power or temperature, limits on fuel burnup, and limits on the leakage of fission products in the reactor cooling system. The design limits shall reflect the importance of preserving the fuel matrix and cladding, as these are first and second barriers to fission product release, respectively. The design shall account for all known degradation mechanisms, with allowance being made for uncertainties in data, calculations, and fuel fabrication. Fuel assemblies shall be designed to permit adequate inspection of their structures and components prior to and following irradiation. In DBAs, the fuel assembly and its component parts shall remain in position with no distortion that would prevent effective post-accident core cooling or interfere with the actions of reactivity control devices or mechanisms. The design shall specify the acceptance criteria necessary to meet these requirements in DBAs. The requirements for reactor and fuel assembly design shall apply in the event of changes in fuel management strategy, or in operating conditions, over the lifetime of the plant. Fuel design and design limits shall reflect a verified and auditable knowledge base. The fuel shall be qualified for operation, either through experience with the same type of fuel in other reactors, or through a program of experimental testing and analysis, to ensure that fuel assembly requirements are met. Guidance The fuel design and qualification should provide assurance that the reactor

core design requirements in section 8.1 are met.

Acceptance criteria should be established for fuel damage, fuel rod failure, and fuel coolability. These criteria should be derived from experiments that identify the limitations of the material properties of the fuel and fuel assembly, and related analyses. The fuel design criteria and other design considerations are discussed below.

Fuel damage

Fuel damage criteria should be established for all known damage mechanisms in operational states (normal operation and AOOs). The damage criteria should assure that fuel dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. The criteria should include stress, strain or loading limits, the cumulative number of strain fatigue cycles, fretting wear, oxidation, hydriding (deuteriding in CANDU reactors), build-up of corrosion products, dimensional changes, rod internal gas pressures, worst-case hydraulic loads, and LWR control rod insertability.

Fuel rod failure

Fuel rod failure applies to operational states, DBAs and DECs. Fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. The design should ensure that fuel does not fail as a result of specific causes during operational states. Fuel rod failures could occur during DBAs and DECs, and are accounted for in the safety analysis.

Assessment methods should be stated for, fuel failure mechanisms, reactor loading and power manoeuvring limitations, and fuel duty which lead to an acceptably low probability of failure. When applicable, the fuel rod failure criteria should consider high burnup effects, based on data of irradiated material properties. The criteria should include:

- hydriding
- cladding collapse
- cladding overheating
- fuel pellet overheating
- excessive fuel enthalpy
- pellet-clad interaction
- stress-corrosion cracking
- cladding bursting
- mechanical fracturing

Fuel coolability

Fuel coolability applies to DBAs and, to the extent practicable, DECs. Fuel coolability criteria should be provided for all damage mechanisms in DBAs

	and DECs. The fuel should be designed to ensure that fuel rod damage will not interfere with effective emergency core cooling. The cladding temperatures should not reach a temperature high enough to allow a significant metal- water reaction to occur, thereby minimizing the potential for fission product release. The criteria should include cladding embrittlement, fuel rod ballooning, structural deformation and, in CANDU, beryllium braze penetration. Other considerations The design should also include: • all expected fuel handling activities • the effects of post-irradiation fuel assembly handling • cooling flow of other components of LWR fuel assembly (such as control rods, poison rods, instrumentation, or neutron sources) Testing, inspection, and surveillance programs Programs for testing and inspection of new fuel as well as for online fuel monitoring and post- irradiation surveillance of irradiated fuel should be established. Fuel specification The design should establish the specification of fuel rods and assembly (including LWR control rods) in order to minimize design deviations and to determine whether all design bases are met (such as limits and tolerances). Reactor core thermal hydraulic design The thermalhydraulic design should be such that sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the reactor coolant system, to prevent fuel sheath overheating. The design requirements can be demonstrated by meeting a set of derived acceptance criteria, as required by REGDOC-2.4.1, Deterministic Safety Analysis. Critical heat flux (CHF) is defined as the heat flux at departure from nucleate boiling (DNB), commonly used in pressurized water reactors (PWRs), or at dryout, commonly used in CANDU designs. It should be noted that, although a thermal margin criterion is sufficient to demonstrate that overheating from a deficient cooling mechanism can be avoided; other mechanistic methods may be acceptable as CHF is not considered as a failure mechanism. In some designs, CHF conditions during transients can be tolerated if it can be shown by other methods that the sheath temperatures do not exceed well-defined acceptable limits. However, any other criteria than the CHF criterion should address sheath

	temperature, pressure, time duration, oxidation, embrittlement etc., and these new criteria should be supported by sufficient experimental and analytical evidence. In the absence of such evidence, the core thermal-hydraulic design is expected to demonstrate a thermal margin to CHF. The demonstration of thermal margin is expected to be presented in a manner that accounts for all possible reactor operational states and conditions, as determined from operating maps including all AOOs. The demonstration should also include long term effects of plant aging and other expected changes to core configuration over the operating life of the plant. The demonstration of thermal margin should thoroughly address uncertainties of various parameters affecting the thermal margin. The design should identify all sources of significant uncertainties that contribute to the uncertainty of thermal margin. The uncertainty for each of the sources should be quantified with supportable evidence. In addition to the demonstration of thermal margin, the core thermal-hydraulic design should also address possible core power and flow oscillations and thermal-hydraulic instabilities. The design should be such that power and flow oscillations that result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.
Macro-Gap	SF01-01-16
Issue/Gap Description	Qualitative acceptance criteria have been established to assess fuel and fuel channel integrity fitness-for-service (FFS) following an AOO. The COG report “The AOO Fuel and Pressure Tube Fitness-For-Service Criteria for LOF, SLOCA and Slow LORC” [COG-12-2049/CG402-RP-001 R01] assesses fuel and fuel channel behaviour during an AOO event. Demonstration that fuel will remain fit for service after AOO cannot be confirmed in the current design documentation. Acceptance criteria and corresponding assessments, including inspection requirements, return to service requirements or further assessments are not available; therefore this is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_8.4.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.4.1 Reactor trip parameters
Requirement Assessed	The design authority shall specify derived acceptance criteria for reactor trip parameter effectiveness for all AOOs and DBAs, and shall perform a safety analysis to demonstrate the effectiveness of the means of shutdown. For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all AOOs and DBAs in time to meet the respective derived acceptance criteria. Where a direct trip parameter does not exist for a given credited means, there shall be two diverse trip parameters specified for that means. For all AOOs and DBAs, there shall be at least two diverse trip parameters unless it can be shown that failure to trip will not lead to unacceptable consequences. There shall be no gap in trip coverage within the OLCs for any operating condition (such as power, temperature), taking into account plant aging. This shall be ensured by the provision of additional trip parameters if necessary. A different level of effectiveness may be acceptable for the additional trip parameters. The extent of trip coverage provided by all available parameters shall be documented for the entire spectrum of failures for each set of PIEs. An assessment of the accuracy and the potential failure modes of the trip parameters shall be provided in the design documentation. Guidance The effectiveness of trip parameters should be assessed through safety analysis performed in accordance with REGDOC-2.4.1, Deterministic Safety Analysis. Trip coverage should be demonstrated across the full range of operating states, for all credited shutdown means and all credited trip parameters. Note that the number of credited shutdown means and the number of credited trip parameters can vary with the event, the reactor design, and whether there is a direct trip available. Defining derived acceptance criteria appropriate to a particular design is the responsibility of the design authority. CNSC REGDOC-2.4.1, Deterministic Safety Analysis, provides the requirements. Derived acceptance criteria should be defined separately for AOOs and

	DBAs. The derived acceptance criteria should be set to give an appropriate level of confidence that a fundamental safety function is assured, or that a barrier to fission product release will not fail. The derived acceptance criteria should: • be quantifiable and well understood • account for the fact that the safety analysis is stylized, and the plant condition at the time of the accident may be significantly different from the analyzed state • cover uncertainties in analysis, input plant and analysis parameters as well as code validation Direct trips are the preferred means of actuating a shutdown means, due to their robustness and low dependence on calculational models. Diverse trip parameters measure different physical variables on the reactor, thus providing additional protection against common mode failure. Where it is impracticable to provide full diversity of trip parameters, different measurement locations, different instrument types and different processing computers should be provided. Manual trip is considered an acceptable trip parameter, if the operator has adequate time to initiate the shutdown action following unambiguous indication of the need to perform the action (in accordance with section 8.10.4). It is the responsibility of the design authority to identify and justify those trip parameters that can be considered “direct”. The design authority should also demonstrate that any trip parameters that are a measure of the event, but not a measure of the challenge to acceptance criteria, cannot be “masked” or “blinded” by control system action or other means. Trips that are dependent on a number of measured variables, such as low DNBR (departure from nucleate boiling ratio) trips in PWRs can only be considered direct if all the variables are direct. Guidance on applying the requirements for number and diversity of trip parameters is given in REGDOC-2.4.1, Deterministic Safety Analysis. REGDOC-2.4.1 also provides the minimum expectations for the number of trip parameters. A manual reactor trip can be considered to be equivalent to a trip parameter, if the requirements for crediting operator action from the main control room are met (see section 8.10.4) and the reliability of manual shutdown meets the reliability requirements for an automatic trip.
Macro-Gap	SF01-01-15
Issue/Gap Description	The analysis in Part 3 of the Safety Report is consistent with demonstrating that both redundant shutdown systems are effective independently in shutdown the reactor. With exceptions of few cases, trip coverage maps for the various events demonstrate that two trips are

	available; however, the applicable trips to every event are not identified as direct or indirect trip (Gap). Acceptance criteria are not explicitly specified for AOOs. Further details are presented in the assessment against REGDOC-2.4.1 requirements in Safety Factor 5.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_8.4.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.4.1 Reactor trip parameters
Requirement Assessed	The design authority shall specify derived acceptance criteria for reactor trip parameter effectiveness for all AOOs and DBAs, and shall perform a safety analysis to demonstrate the effectiveness of the means of shutdown. For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all AOOs and DBAs in time to meet the respective derived acceptance criteria. Where a direct trip parameter does not exist for a given credited means, there shall be two diverse trip parameters specified for that means. For all AOOs and DBAs, there shall be at least two diverse trip parameters unless it can be shown that failure to trip will not lead to unacceptable consequences. There shall be no gap in trip coverage within the OLCs for any operating condition (such as power, temperature), taking into account plant aging. This shall be ensured by the provision of additional trip parameters if necessary. A different level of effectiveness may be acceptable for the additional trip parameters. The extent of trip coverage provided by all available parameters shall be documented for the entire spectrum of failures for each set of PIEs. An assessment of the accuracy and the potential failure modes of the trip parameters shall be provided in the design documentation. Guidance The effectiveness of trip parameters should be assessed through safety analysis performed in accordance with REGDOC-2.4.1, Deterministic Safety Analysis. Trip coverage should be demonstrated across the full range of operating states, for all credited shutdown means and all credited trip parameters. Note that the number of credited shutdown means and the number of credited trip parameters can vary with the event, the reactor design, and whether there is a direct trip available. Defining derived acceptance criteria appropriate to a particular design is the responsibility of the design authority. CNSC REGDOC-2.4.1, Deterministic Safety Analysis, provides the requirements. Derived acceptance criteria should be defined separately for AOOs and

	DBAs. The derived acceptance criteria should be set to give an appropriate level of confidence that a fundamental safety function is assured, or that a barrier to fission product release will not fail. The derived acceptance criteria should: • be quantifiable and well understood • account for the fact that the safety analysis is stylized, and the plant condition at the time of the accident may be significantly different from the analyzed state • cover uncertainties in analysis, input plant and analysis parameters as well as code validation Direct trips are the preferred means of actuating a shutdown means, due to their robustness and low dependence on calculational models. Diverse trip parameters measure different physical variables on the reactor, thus providing additional protection against common mode failure. Where it is impracticable to provide full diversity of trip parameters, different measurement locations, different instrument types and different processing computers should be provided. Manual trip is considered an acceptable trip parameter, if the operator has adequate time to initiate the shutdown action following unambiguous indication of the need to perform the action (in accordance with section 8.10.4). It is the responsibility of the design authority to identify and justify those trip parameters that can be considered “direct”. The design authority should also demonstrate that any trip parameters that are a measure of the event, but not a measure of the challenge to acceptance criteria, cannot be “masked” or “blinded” by control system action or other means. Trips that are dependent on a number of measured variables, such as low DNBR (departure from nucleate boiling ratio) trips in PWRs can only be considered direct if all the variables are direct. Guidance on applying the requirements for number and diversity of trip parameters is given in REGDOC-2.4.1, Deterministic Safety Analysis. REGDOC-2.4.1 also provides the minimum expectations for the number of trip parameters. A manual reactor trip can be considered to be equivalent to a trip parameter, if the requirements for crediting operator action from the main control room are met (see section 8.10.4) and the reliability of manual shutdown meets the reliability requirements for an automatic trip.
Macro-Gap	SF01-01-16
Issue/Gap Description	With the exception of a few cases, trip coverage maps for the various events demonstrate that two trips are effective; however, the applicable trips to every event are not identified as direct or indirect trip (Gap). Acceptance criteria are not explicitly specified for AOOs. Further details

	are presented in the assessment against CNSC REGDOC-2.4.1 requirements in Safety Factor 5.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_8.9.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.9.3 Alternate AC power supply
Requirement Assessed	The electrical power system design shall include provisions for mitigating the complete loss of onsite and offsite AC power. This is accomplished by the use of onsite portable, transportable or fixed power sources or offsite portable or transportable power sources, or a combination of these. The alternate AC power source shall be available and located at or nearby the NPP, and shall: 1. be connectable to but not normally connected to the offsite or onsite standby and emergency AC power systems 2. have minimum potential for common mode failure with offsite power or the onsite standby and emergency AC power sources 3. be available in a timely manner after the onset of a station blackout 4. have sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in a safe shutdown state The design shall include provision for periodic capacity testing of the alternate power supply to confirm its capability to cope with a station blackout event. Guidance The plant's capability to maintain critical parameters (reactor coolant inventory, containment temperature and pressure, room temperatures where critical equipment is located) and to remove decay heat from irradiated fuel should be analyzed for the period that the plant is in a station blackout (SBO) condition. The capability of the DC systems required to monitor critical parameters and power the lighting and communication systems during an SBO should be evaluated for adequacy.
Macro-Gap	SF01-12-15
Issue/Gap Description	Provisions for mitigating complete loss of onsite and offsite AC power have not been considered in the original design of Bruce A electrical power systems. Since the heat transport system pumps are one of the major unit Class IV system loads. Failures in the Class IV power system can result in a loss of power to one or more of these pumps, with a consequent reduction of forced circulation in the heat transport system. The safety concerns associated with such events are possible impairment of fuel cooling capability and pressurization of the heat transport system which

	may pose a threat to the integrity of the heat transport system. Analysis of a number of postulated failures in the Class IV power system, leading to either total or partial loss of Class IV power to a unit is performed to demonstrate the capability of the design to accommodate such failures. The current safety analysis as documented in Part 3 of the Safety Report does not consider events with station blackout. Therefore, this is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_8.9.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.9.3 Alternate AC power supply
Requirement Assessed	The electrical power system design shall include provisions for mitigating the complete loss of onsite and offsite AC power. This is accomplished by the use of onsite portable, transportable or fixed power sources or offsite portable or transportable power sources, or a combination of these. The alternate AC power source shall be available and located at or nearby the NPP, and shall: 1. be connectable to but not normally connected to the offsite or onsite standby and emergency AC power systems 2. have minimum potential for common mode failure with offsite power or the onsite standby and emergency AC power sources 3. be available in a timely manner after the onset of a station blackout 4. have sufficient capacity and reliability for operation of all systems required for coping with station blackout and for the time required to bring and maintain the plant in a safe shutdown state The design shall include provision for periodic capacity testing of the alternate power supply to confirm its capability to cope with a station blackout event. Guidance The plant's capability to maintain critical parameters (reactor coolant inventory, containment temperature and pressure, room temperatures where critical equipment is located) and to remove decay heat from irradiated fuel should be analyzed for the period that the plant is in a station blackout (SBO) condition. The capability of the DC systems required to monitor critical parameters and power the lighting and communication systems during an SBO should be evaluated for adequacy.
Macro-Gap	SF01-12-16
Issue/Gap Description	The current safety analysis as documented in Part 3 of the Safety Report does not consider events with station blackout. Therefore, this is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_9.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.1 General
Requirement Assessed	A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals. Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included. The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.
Macro-Gap	SF01-13-15
Issue/Gap Description	The radioactive sources other than the reactor core are not addressed in Part 3 of the Safety Report. A limited set of Fuel Handling System Failures is discussed in Appendix 1 Section 1.5 of Part 3 of the Safety Report. Therefore, it is assessed as a gap (Gap).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_9.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.1 General
Requirement Assessed	A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals. Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included. The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.
Macro-Gap	SF01-13-16
Issue/Gap Description	The radioactive sources other than the reactor core are not addressed in Part 3 of the Safety Report. A limited set of Fuel Handling System Failures is discussed in Appendix 1 Section 1.5 of Part 3 of the Safety Report. Therefore, it is assessed as a gap (Gap 1).
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_9.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.1 General
Requirement Assessed	A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals. Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included. The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.
Macro-Gap	SF01-01-16
Issue/Gap Description	As for clause 7.4, systematic methodology for event identification is not demonstrated (Gap 2)
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_9.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.2 Analysis objectives
Requirement Assessed	The safety analysis shall be iterative with the design process, and result in two reports: a preliminary safety analysis report, and a final safety analysis report. The preliminary safety analysis shall assist in the establishment of the design-basis requirements for the items important to safety, and demonstrate whether the plant design meets applicable requirements. The final safety analysis shall: 1. reflect the as-built plant 2. account for postulated aging effects on SSCs important to safety 3. demonstrate that the design can withstand and effectively respond to identified PIEs 4. demonstrate the effectiveness of the safety systems and safety support systems 5. derive the OLCs for the plant, including: a. operational limits and set points important to safety b. allowable operating configurations, and constraints for operational procedures 6. establish requirements for emergency response and accident management 7. determine post-accident environmental conditions, including radiation fields and worker doses, to confirm that operators are able to carry out the actions credited in the analysis 8. demonstrate that the design incorporates sufficient safety margins 9. confirm that the dose and derived acceptance criteria are met for all AOOs and DBAs 10. demonstrate that all safety goals have been met Guidance The Class I Nuclear Facilities Regulations requires a preliminary safety analysis report demonstrating the adequacy of the NPP design to be submitted in support of an application for a licence to construct a Class I nuclear facility. A final safety analysis report demonstrating the adequacy of the design is required for an application for a licence to operate a Class I nuclear facility.
Macro-Gap	SF01-01-15
Issue/Gap Description	The main gap is that AOOs acceptance criteria are not assessed separately since AOOs are not identified explicitly (Gap). For more details see assessment against REGDOC-2.4.1 requirements. Further details are presented in Safety Factor 5.
Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_9.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.2 Analysis objectives
Requirement Assessed	The safety analysis shall be iterative with the design process, and result in two reports: a preliminary safety analysis report, and a final safety analysis report. The preliminary safety analysis shall assist in the establishment of the design-basis requirements for the items important to safety, and demonstrate whether the plant design meets applicable requirements. The final safety analysis shall: 1. reflect the as-built plant 2. account for postulated aging effects on SSCs important to safety 3. demonstrate that the design can withstand and effectively respond to identified PIEs 4. demonstrate the effectiveness of the safety systems and safety support systems 5. derive the OLCs for the plant, including: a. operational limits and set points important to safety b. allowable operating configurations, and constraints for operational procedures 6. establish requirements for emergency response and accident management 7. determine post-accident environmental conditions, including radiation fields and worker doses, to confirm that operators are able to carry out the actions credited in the analysis 8. demonstrate that the design incorporates sufficient safety margins 9. confirm that the dose and derived acceptance criteria are met for all AOOs and DBAs 10. demonstrate that all safety goals have been met Guidance The Class I Nuclear Facilities Regulations requires a preliminary safety analysis report demonstrating the adequacy of the NPP design to be submitted in support of an application for a licence to construct a Class I nuclear facility. A final safety analysis report demonstrating the adequacy of the design is required for an application for a licence to operate a Class I nuclear facility.
Macro-Gap	SF01-01-16
Issue/Gap Description	The main gap is that AOOs acceptance criteria are not assessed separately since AOOs are not identified explicitly (Gap). For more details see assessment against REGDOC-2.4.1 requirements. Further details are presented in Safety Factor 5.

Rationale	See Notes 1 and 2

Gap #	SF01_CNSC REGDOC 2.5.2_9.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.3 Hazard analysis
Requirement Assessed	Hazard analysis shall collect and evaluate information about the NPP to identify the associated hazards and determine those that are significant and must be addressed. A hazard analysis shall demonstrate the ability of the design to effectively respond to credible common-cause events. As discussed in section 9.1, the first step of the hazard analysis is to identify PIEs. For each common-cause PIE, the hazard analysis shall identify: 1. applicable acceptance criteria (i.e., the success path criteria) 2. the hazardous materials in the plant and at the plant site 3. all qualified mitigating SSCs credited during and following the event all non-qualified safety or safety support systems are assumed to fail, except in cases where their continued operation would result in more severe consequences 4. operator actions and operating procedures for the event 5. plant or operating procedure parameters for which the event is limiting The hazard analysis shall confirm that: 1. the plant design incorporates sufficient diversity and separation to cope with credible common-cause events 2. credited SSCs are qualified to survive and function during and following credible common- cause events, as applicable 3. the following criteria are met: a. the plant can be brought to a safe shutdown state b. the integrity of the fuel in the reactor core can be maintained c. the integrity of the reactor coolant pressure boundary and containment can be maintained d. safety-critical parameters can be monitored by the operator The hazard analysis report shall include the findings of the analysis and the basis for those findings. This report shall also: 1. include a general description of the physical characteristics of the plant that outlines the prevention and protection systems to be provided

	2. include the list of safe shutdown equipment 3. define and describe the characteristics associated with hazards for all areas that contain hazardous materials 4. describe the performance criteria for detection systems, alarm systems, and mitigation systems, including requirements such as seismic or environmental qualification 5. describe the control and operating room areas and the protection systems provided for these areas, including additional facilities for maintenance and operating personnel 6. describe the operator actions and operating procedures of importance to the given analysis 7. identify the plant parameters for which the event is limiting 8. explain the inspection, testing, and maintenance parameters needed to protect system integrity 9. define the emergency planning and coordination requirements for effective mitigation, including any necessary measures to compensate for the failure or inoperability of any active or passive protection system or feature Guidance The objective of the hazard analysis is to determine the adequacy of protection of the NPP against internal and external hazards, while taking into account the plant design and site characteristics. To ensure the availability of required safety functions and operator actions, all the SSCs important to safety (including the main control room, secondary control room and emergency support facilities) should be adequately protected against relevant internal and external hazards. The hazard analysis should establish a list of relevant internal and external hazards that may affect plant safety. For the relevant hazards, the review should demonstrate, by using deterministic and probabilistic techniques, that the probability or consequences of the hazard are sufficiently low so that no specific protective measures are necessary, or that the preventive and mitigating measures against the hazard are adequate. All internal and external hazards are considered as part of PIEs. The hazards that make an insignificant contribution to plant risk can be screened out from the detailed analysis; however, the rationale for this screening should be provided. The remaining PIEs constitute the scope of the hazard analysis. The design should specify design-basis hazards, establishing clear criteria. The design-basis hazards should be analyzed using the deterministic safety analysis rules and criteria

	provided in section 9.4. Such analysis should also demonstrate the adequacy of the complementary design features in mitigating radiological consequences of design extension conditions. The hazard analysis should demonstrate that the design incorporates sufficient safety margins.
Macro-Gap	SF01-09-15
Issue/Gap Description	In regards to point 4, the manual actions credited in the Fire Safe Shutdown Assessment have not been identified in operating procedures (Gap). These procedures must be developed or updated to incorporate these operator actions. As a result of the improvements of fire protection provisions to achieve alignment with N293-07 requirements and to follow up from the Bruce A FSSA that specified Operator actions that are potentially required to meet the station fire safe shutdown goals for some of the postulated fires, Bruce Power will conduct a review of Bruce A Operator manual actions. This review will assess if any gaps exist in the required response to hazards identified in the FSSA. This review has already been conducted at Bruce B and determined that no gaps exist. This action is monitored under AI 1207-3890.
Rationale	See Notes 1 and 2

Gap #	SF01_CSA N290.1_4.3.1.4_16
Document ID	CSA N290.1
Article/Clause	4.3.1.4
Requirement Assessed	In order to credit (in the safety analysis) operator action to shut down (manually trip) the reactor, the design shall provide a) clear, well-defined, validated, and readily available operating procedures that identify the necessary actions; b) instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action; c) adequate time before operator action is required, following indication of the necessity for operator action inside the control rooms; and d) adequate time before operator action is required, following indication of the necessity for operator action outside the control rooms. Notes: 1) For new plants, adequate time is at least 30 min for operator action inside the control room and 60 min for operator action outside the control room. 2) For existing CANDU plants, adequate time is 15 min for operator action inside the control room and 30 min for operator action outside the control room.
Macro-Gap	SF01-09-16
Issue/Gap Description	Part (c) this clause (CSA N290.1 c. 4.3.1.4) is not met. As identified in Table 1-3, analysis of HTS depressurization due to steam bleed valves open with pressurizer heaters off and multiple failures of the bleed condenser to isolate, operator action to manually trip the reactor was credited at 12 minutes in order to prevent sheath dryout (see Appendix 3, Section 3.5.4.2 of Part 3 of the Safety Report). This is not considered adequate time (defined as 15 minutes per note 2) and is therefore considered a gap (Gap 1).
Rationale	See Note 2

Gap #	SF04_CSA N285.4-14_12.5_15
Document ID	CSA N285.4-14
Article/Clause	12.5 Material surveillance of fuel channel annulus spacers
Requirement Assessed	Note: The measurements specified in Clause 12.5 are intended to monitor fuel channel annulus spacer material properties. The requirements are defined for spacers manufactured in accordance with CSA N285.6.5-Series-88 or N285.6.10-12.
Macro-Gap	SF04-02-15
Issue/Gap Description	The specific requirements in N285.4-14 on monitoring of fuel channel annulus spacer material properties will need to be addressed if Bruce Power is required to comply with this version of the standard in the future. Consideration should be given to developing guidance for monitoring annual spacer material properties
Rationale	B-PLAN-31100-00001 Fuel Channel Life Cycle Management Plan Section 4.4.3.3 Spacer Degradation addresses monitoring of fuel channel annulus spacer material properties through participation in COG JP-4363. Letter from F. Saunders to M. Leblanc, ' Bruce A and Bruce B Licence Renewal-Supplemental Update', dated, November 27, 2014, NK21-CORR-00531-11711, NK29-CORR-00531-12101: Page A14 & A15 of A47 states the following: The material properties of loose-fitting spacers (Zr-Nb-Cu material), which are installed in most channels in Bruce Units 3 to 7, are not considered a concern based on OPEX and testing ex-service material. Tight-fitting spacers (Inconel X-750 material) are used for all spacers in Bruce Units 1, 2 and 8, and in limited number of channels in Bruce Units 3 to 7. Testing of tight-fitting Inconel X-750 spacer condition through testing of spacers from removed pressure tubes, along with research and development and modeling activities, will be used to demonstrate fitness-for-service for continued reactor operation. Bruce Power is planning to remove its next surveillance tube from Unit 8 in order to obtain more information on Inconel X-750 spacer degradation. Hence, this potential improvement is considered to be In-Progress as part of the FC life cycle management activities.

Gap #	SF04_CSA N285.4-14_12.5_16
Document ID	CSA N285.4-14
Article/Clause	12.5
Requirement Assessed	Material surveillance of fuel channel annulus spacers This clause requires the licensee to prepare an annulus spacer material surveillance program. Additional requirements covered by this clause include extent of testing and sample size, spacer testing intervals, measurement methods and procedures, evaluation of results and dispositions, and records.
Macro-Gap	SF04-02-16
Issue/Gap Description	The specific requirements in CSA-N285.4-14 on monitoring of fuel channel annulus spacer material properties is not addressed.
Rationale	B-PLAN-31100-00001 Fuel Channel Life Cycle Management Plan Section 4.4.3.3 Spacer Degradation addresses monitoring of fuel channel annulus spacer material properties through participation in COG JP-4363. Letter from F. Saunders to M. Leblanc, ' Bruce A and Bruce B Licence Renewal-Supplemental Update', dated, November 27, 2014, NK21-CORR-00531-11711, NK29-CORR-00531-12101: Page A14 & A15 of A47 states the following: The material properties of loose-fitting spacers (Zr-Nb-Cu material), which are installed in most channels in Bruce Units 3 to 7, are not considered a concern based on OPEX and testing ex-service material. Tight-fitting spacers (Inconel X-750 material) are used for all spacers in Bruce Units 1, 2 and 8, and in limited number of channels in Bruce Units 3 to 7. Testing of tight-fitting Inconel X-750 spacer condition through testing of spacers from removed pressure tubes, along with research and development and modeling activities, will be used to demonstrate fitness-for-service for continued reactor operation. Bruce Power has removed B8J18 from service and testings on the retrieved Inconel X-750 spacers are currently in progress. Fitness-for-service assessments will be performed based on the testing results. Hence, this potential improvement is considered to be In-Progress as part of the FC life cycle management activities.

Gap #	SF05_CNSC REGDOC 2.3.2_3.4_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	3.4 Requirements for procedures and guidelines
Requirement Assessed	Licensees shall: 1. develop, verify and validate accident management procedures and guidelines, including EOPs and SAMGs 2. account for factors specific to the reactor design in the development of SAMGs for severe accidents 3. consider that information available to the operating staff or emergency groups may be incomplete and characterized by significant uncertainties 4. include the following in SAMGs: a. the parameters and their thresholds that define the transition from EOPs to SAMGs b. key parameters to diagnose the state of various reactor and reactor systems throughout the progression of the accident c. actions to be taken to counter the damage mechanisms that would potentially challenge the integrity of the containment, irrespective of predicted frequencies of occurrence for those damage mechanisms d. indicators that can be used to judge the success of the implemented actions e. the communication protocol to be followed during implementation of accident management f. guidance on dealing with multi-unit damage, uncovered fuel in spent fuel pools, releases of radioactive materials and hydrogen into buildings adjacent to the containment 5. ensure the EOPs and SAMGs consider sufficiently long time periods to initiate and complete required actions, taking into account the human and organizational performance and the possibility of prolonged time required to restore power due to multi-unit damage or large-scale external disturbances 6. include necessary steps into guidelines for events where supplementary equipment (also called emergency mitigating equipment (EME)) and where external supports are required to mitigate the accident consequences 7. provide for transition from the accident management activities to accident recovery
Macro-Gap	SF05-06-15
Issue/Gap Description	A comprehensive set of Bruce Power specific AIMs and SAMG procedures are prepared. The technical basis, entry and exit conditions, and assumptions used in AIM procedures make use of the deterministic analysis of the design basis events, while those used in SAMG technical basis are largely based on the deterministic safety analysis of severe BDBAs analyzed within PRA Level 2 scope as well as PRA Level 1 and 2. Significant progress has been made on a large number of planned post Fukushima design enhancements to prevent and mitigate severe accidents. These enhancements include, e.g., adding design features to

	allow external water makeup to the HTS, moderator system, steam generators and the irradiated fuel bay as well as enhancements to the emergency power supply. PRA assessments that take into account Emergency Mitigation Equipment demonstrate significant improvements in SCDF and releases (see SFR 6 for details). The SAMG was developed to guide response to a severe accident occurring on a single unit only. For multi-unit sites, Bruce A PRA indicates that multi-unit events are considered. The completeness of such consideration needs to be confirmed, in particular, it may require complementary DSA for BDBAs to confirm meeting the safety goals (Gap 1).
Rationale	Complementary DSA for BDBAs are covered under Safety Report Improvement Project (SIRP). See Note 2.

Gap #	SF05_CNSC REGDOC 2.3.2_4.2.1_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	4.2.1 Identification of challenges to reactor safety functions
Requirement Assessed	The development of an IAMP should consider postulated initiating events and accident sequences that could be caused by credible failures or malfunctions of SSCs, human errors, common-cause internal and external hazards, and combinations thereof. Challenges that are not considered in the reactor design envelope, but could potentially threaten the integrity of the containment should be practically eliminated; that is, the existing process systems, safety and control systems, complementary design features, available SSCs, and procedural provisions should make the occurrence of these challenges practically impossible. For example, the installed rupture disks or relief valves that provide reliable and sufficient depressurization capability for a reactor core or vessel can eliminate the high-pressure corium ejection phenomenon and thus the possibility of direct containment heating by corium. Among credible events, a selected set of accident sequences that can be used to represent the consequences of each group of accident sequences should be used to obtain insights into the behaviour of the accident and to identify challenges to reactor safety functions. This requires investigating how specific accidents will challenge safety functions and – if safety functions are lost and not restored in due time – how the accident progresses, how the fission product barriers are breached, how long it will take to reach each stage of the accident, and how severe each accident stage will be. In the domain of beyond-design-basis accidents (BDBA), insights into the response of the reactor to BDBAs, including severe accidents, should be obtained. A technical basis for SAM should document the understanding of severe accident phenomena and reactor-specific physical processes, such as core degradation, in-vessel core debris retention, ex-vessel corium spreading and coolability, molten fuel coolant interaction, molten core concrete interaction, and all known containment challenge mechanisms. The technical basis should also include severe accident phenomena in spent fuel bays and multi-unit distress. The technical basis should be updated as necessary to reflect the state-of-the- art knowledge and experimental data obtained from applicable severe accident research programs and lessons learned from the reactors that have experienced severe core damage. The updated knowledge and data should be used to evaluate the reactor ability to cope with accidents and to deduce suitable accident management strategies, provisions, procedures, and guidelines. Reactor-specific beyond-design-basis initiating events, such as events triggered by extreme external hazards (e.g., earthquakes, flooding, and

	extreme weather conditions), should also be considered to increase the reactor coping capability. The aim is to ensure that a set of sufficient, supplementary onsite equipment and consumables (e.g., fuel and water inventories) are identified, obtained, protected and stored onsite or offsite. These can be used to maintain or restore the cooling of the core, the containment, and the spent fuel pool following a beyond-design-basis initiating event. After the consumables are used up, offsite resources should be obtained to sustain those cooling functions indefinitely. Accident management should consider that some beyond-design-basis initiating events may result in similar challenges to all units on the site. Challenges for severe accidents and beyond-design-basis initiating events may be identified using a targeted assessment of safety margins against a set of postulated extreme conditions that cause a consequential loss of safety functions leading to severe core damage. Such a reactor-specific “stress test” can be used to determine the time of autonomy of reactor-critical safety functions, any potential weak points, and any cliff-edge effects for a given set of the considered extreme situations. This type of exercise may be used to identify the potential for safety improvements and to provide input to the development of an IAMP.
Macro-Gap	SF05-06-15
Issue/Gap Description	The SAMG was developed to guide response to a severe accident occurring on a single unit only. Bruce A PRA indicates that multi-unit events are considered. The completeness of such consideration needs to be confirmed, in particular, it may require complementary DSA for BDBAs to confirm meeting the safety goals (Gap 1).
Rationale	If required, complementary DSA for BDBAs are covered under Safety Report Improvement Project (SIRP). See Note 2.

Gap #	SF05_CNSC REGDOC 2.3.2_4.2.5_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	4.2.5 Development of procedures and guidelines
Requirement Assessed	Procedures and guidelines to implement the strategies and measures for accident management should be developed and described in documents such as EOPs and SAMGs, or equivalent documents (see the requirements specified in section 3.4). If EOPs and SAMGs already exist, the IAMP can be built using these existing elements. Any new information on reactor site configuration, changes in hazards, and knowledge gained should be considered, and if appropriate procedures and guidelines should be updated accordingly. The EOPs should contain a set of information, instructions and actions designed to prevent the escalation of an accident, mitigate its consequences and bring the reactor to a safe and stable state. The SAMGs should contain a set of information, instructions and actions designed to mitigate the consequences of a severe accident according to the chosen strategies. Uncertainties may exist both in the reactor status and in the outcome of a selected action. Therefore, SAMGs should propose a range of possible actions and allow for additional evaluation and alternative actions. SAMGs should also address various positive and negative consequences of proposed actions, including the use of equipment, limitations of the equipment, cautions and benefits. The procedures and guidelines should be verified and validated. This should include the usability of the procedures and guidelines (see section 5.2). Clear criteria for EOP to SAMG transition should be defined. Adequate guidance should be provided in the design of the IAMP to ensure that its event and symptom-based EOP components, or equivalent, are appropriately coordinated among the responsible personnel and that the symptom-based approach is invoked when it is required. Measures, including providing guidelines and training, should be defined to support staff decision- making for situations where an event has progressed to a stage for which procedures have not been defined. EOPs and SAMGs should cover events with multi-unit damage, potential damage to the fuel in spent fuel pools, releases of radioactive materials and hydrogen into buildings adjacent to the containment, and run-off of contaminated water to the environment. The time period that EOPs or SAMGs assume to initiate and complete required actions should reflect potential damage to the reactor. For

	example, a SAMG may specify a time period required to hook up alternative power and water sources. For external events, the extent of reactor damage and disturbances from outside or at the grid should be taken into account to prolong this time period. Having a diesel back on line may take a whole day or even longer, much more than the time that is assumed sufficient for an intact site area without large disturbances from outside. For beyond-design-basis initiating events, the reactor may require supplementary equipment stored onsite or offsite and external support to mitigate the accident consequences. These necessary measures should be specified in guidelines for coping with these events.
Macro-Gap	SF05-06-15
Issue/Gap Description	There is a gap against this clause requirement relevant to events with multi-unit. As identified against previous clauses (Gap 1).
Rationale	If required, complementary DSA for BDBAs are covered under Safety Report Improvement Project (SIRP). See Note 2.

Gap #	SF05_CNSC REGDOC 2.4.1_3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	3 Objectives
Requirement Assessed	Safety analysis is an essential element of a safety assessment. It is an analytical study used to demonstrate how safety requirements are met for a broad range of operating conditions and various initiating events. Safety analysis involves deterministic and probabilistic analyses in support of the siting, design, commissioning, operation or decommissioning of an NPP. This document focuses on the deterministic safety analysis used in the evaluation of event consequences. PSA and hazard analysis are outside the scope of this document – the requirements for probabilistic safety assessments for NPPs are provided in regulatory document REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants (formerly S-294). The objectives of deterministic analysis are to: 1. confirm that the design of an NPP meets design and safety requirements 2. derive or confirm operational limits and conditions that are consistent with the design and safety requirements for the NPP 3. assist in establishing and validating accident management procedures and guidelines 4. assist in demonstrating that safety goals, which may be established to limit the risks posed by the NPP, are met This document identifies high-level requirements for conducting and presenting a safety analysis, taking into account best national and international practices.
Macro-Gap	SF05-11-15
Issue/Gap Description	1. Safety analysis of the effectiveness of the special safety systems and the applicable alternative heat sink systems was performed and is documented in the SR. Some other analyses in support of design and operation would be documented external to the SR, however, stress analysis for Bruce A shield cooling system is not performed to confirm the design and safety requirement (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	3 Objectives
Requirement Assessed	Safety analysis is an essential element of a safety assessment. It is an analytical study used to demonstrate how safety requirements are met for a broad range of operating conditions and various initiating events. Safety analysis involves deterministic and probabilistic analyses in support of the siting, design, commissioning, operation or decommissioning of an NPP. This document focuses on the deterministic safety analysis used in the evaluation of event consequences. PSA and hazard analysis are outside the scope of this document – the requirements for probabilistic safety assessments for NPPs are provided in regulatory document REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants (formerly S-294). The objectives of deterministic analysis are to: 1. confirm that the design of an NPP meets design and safety requirements 2. derive or confirm operational limits and conditions that are consistent with the design and safety requirements for the NPP 3. assist in establishing and validating accident management procedures and guidelines 4. assist in demonstrating that safety goals, which may be established to limit the risks posed by the NPP, are met This document identifies high-level requirements for conducting and presenting a safety analysis, taking into account best national and international practices.
Macro-Gap	SF05-01-15
Issue/Gap Description	2. Analyses of some events establishing OLCs were done pre-2001 using legacy codes (Gap 2).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.1 Responsibilities
Requirement Assessed	The licensee is responsible for ensuring that the safety analysis meets all regulatory requirements. The licensee shall: 1. maintain adequate capability to perform or procure safety analysis 2. establish a formal process to assess and update safety analysis, which takes into account operational experience, research findings and identified safety issues 3. establish and apply a formal quality assurance (QA) process that meets the QA standards established for safety analysis in CSA Group N286.7-99, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants Guidance As stated in this regulatory document, the licensee must maintain adequate capability to perform or procure safety analysis in order to: • resolve technical issues that arise over the life of the plant • ensure the safety analysis requirements are met for the safety analysis developed by the operating organization or procured from a third party A formal process should be established to assess and update the safety analysis to ensure that the safety analysis reflects: • current plant configuration (for existing plants) • current operating limits and conditions (for existing plants) • operating experience, including the experience from similar facilities • results available from experimental research, improved theoretical understanding or new modelling capabilities to assess potential impacts on the conclusions of safety analyses • human factors considerations, to ensure that credible estimates of human performance are used in the analysis
Macro-Gap	SF05-01-15
Issue/Gap Description	[DPT-NSAS-00011] on Configuration Management of Safety Analysis Software was prepared in consideration of N286.7-99. Although Bruce Power does not perform development or maintenance activities of the safety analysis software, it has acquired the right to use these computer codes from the Hosting Organizations by multiparty or bilateral agreements. As such, this procedure is limited to the description of the processes for use of safety analysis software, requesting software

	changes to the owner organizations and modification to scripts and utility codes. However, a number of the legacy analyses in the Safety Report are performed with codes (including the models and data) that have not been verified and validated to the requirements of CSA N286.7-99 and therefore is considered a gap against this requirement (Gap 1)
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.1 Responsibilities
Requirement Assessed	The licensee is responsible for ensuring that the safety analysis meets all regulatory requirements. The licensee shall: 1. maintain adequate capability to perform or procure safety analysis 2. establish a formal process to assess and update safety analysis, which takes into account operational experience, research findings and identified safety issues 3. establish and apply a formal quality assurance (QA) process that meets the QA standards established for safety analysis in CSA Group N286.7-99, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants Guidance As stated in this regulatory document, the licensee must maintain adequate capability to perform or procure safety analysis in order to: • resolve technical issues that arise over the life of the plant • ensure the safety analysis requirements are met for the safety analysis developed by the operating organization or procured from a third party A formal process should be established to assess and update the safety analysis to ensure that the safety analysis reflects: • current plant configuration (for existing plants) • current operating limits and conditions (for existing plants) • operating experience, including the experience from similar facilities • results available from experimental research, improved theoretical understanding or new modelling capabilities to assess potential impacts on the conclusions of safety analyses • human factors considerations, to ensure that credible estimates of human performance are used in the analysis
Macro-Gap	SF05-01-16
Issue/Gap Description	A number of the legacy analyses in the Safety Report are performed with codes (including the models and data) that have not been verified and validated to the requirements of CSA N286.7-99.
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.1 Identification of events
Requirement Assessed	The licensee shall use a systematic process to identify events, event sequences, and event combinations ("events" hereafter in this document) that can potentially challenge the safety or control functions of the NPP. The licensee shall also identify events that may lead to fission product releases, including those related to spent fuel pools (also called irradiated fuel bays) and fuel-handling systems. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design. The identification of events will include at-power and shutdown states. The deterministic analysis should also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown analysis. Common-cause events affecting multiple reactor units on a site shall be considered. The list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. In addition to events that could challenge the safety or control functions of the NPP, safety analysis shall be performed for normal operation. Guidance The safety analysis is performed for a set of events that could lead to challenges related to the NPP's safety or control functions. These include events caused by SSC failures or human error as well as human-induced or natural common-cause events. The events considered in safety analysis could be single PIEs, sequences of several consequential events, or combinations of independent events. The set of events to be considered in safety analysis is identified using a systematic process and by taking into account: • reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams • lists of events developed for safety analysis of other NPPs, as applicable • analysis of operating experience data for similar plants • any events prescribed for inclusion in safety analysis by regulatory requirements (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants) • equipment failures, human errors and common-cause events

	identified iteratively with PSA • a cut-off frequency for common-cause events that is consistent across all events The list of identified events should be iteratively reviewed for accuracy and completeness as the plant design and safety analyses proceed. Reviews should also be periodically conducted throughout the NPP lifecycle, to account for new information and requirements. This regulatory document requires that, when identifying events, all permissible plant operating modes be considered. All operating modes used for extended periods of time should be analyzed. Modes that occur transiently or briefly can be addressed without a specific analysis, as long as it can be shown that existing safety analyses bound the behaviour and consequences of those states. NPP operating modes include, but are not limited to: • initial approach to reactor criticality • reactor start-up from shutdown through criticality to power • steady-state power operation, including both full and low power • changes in the reactor power level, including load follow modes (if employed) • reactor shutting down from power operation • shutdown in a hot standby mode • shutdown in a cold shutdown mode • shutdown in a refuelling mode or maintenance mode that opens major closures in the reactor coolant pressure boundary • shutdown in other modes or plant configurations with unique temperature, pressure or coolant inventory conditions • operation of limited duration, with some systems important to safety being unavailable For events identified by the systematic process used for this purpose, a full range of configurations and operating modes of equipment should be considered in the deterministic safety analysis. Special plant configurations may occur during major plant modifications such as plant refurbishment, lay-up, or decommissioning. These configurations should be considered, and potential events should be identified and included in the deterministic safety analysis.
Macro-Gap	SF05-02-15
Issue/Gap Description	Natural common cause events are not addressed in the Safety Report (Gap 1). It is being considered in the first phase of REGDOC-2.4.1 implementation.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.1 Identification of events
Requirement Assessed	The licensee shall use a systematic process to identify events, event sequences, and event combinations ("events" hereafter in this document) that can potentially challenge the safety or control functions of the NPP. The licensee shall also identify events that may lead to fission product releases, including those related to spent fuel pools (also called irradiated fuel bays) and fuel-handling systems. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design. The identification of events will include at-power and shutdown states. The deterministic analysis should also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown analysis. Common-cause events affecting multiple reactor units on a site shall be considered. The list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. In addition to events that could challenge the safety or control functions of the NPP, safety analysis shall be performed for normal operation. Guidance The safety analysis is performed for a set of events that could lead to challenges related to the NPP's safety or control functions. These include events caused by SSC failures or human error as well as human-induced or natural common-cause events. The events considered in safety analysis could be single PIEs, sequences of several consequential events, or combinations of independent events. The set of events to be considered in safety analysis is identified using a systematic process and by taking into account: • reviews of the plant design using such methods as hazard and operability analysis, failure mode and effects analysis, and master logic diagrams • lists of events developed for safety analysis of other NPPs, as applicable • analysis of operating experience data for similar plants • any events prescribed for inclusion in safety analysis by regulatory requirements (e.g., REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants) • equipment failures, human errors and common-cause events

	identified iteratively with PSA • a cut-off frequency for common-cause events that is consistent across all events The list of identified events should be iteratively reviewed for accuracy and completeness as the plant design and safety analyses proceed. Reviews should also be periodically conducted throughout the NPP lifecycle, to account for new information and requirements. This regulatory document requires that, when identifying events, all permissible plant operating modes be considered. All operating modes used for extended periods of time should be analyzed. Modes that occur transiently or briefly can be addressed without a specific analysis, as long as it can be shown that existing safety analyses bound the behaviour and consequences of those states. NPP operating modes include, but are not limited to: • initial approach to reactor criticality • reactor start-up from shutdown through criticality to power • steady-state power operation, including both full and low power • changes in the reactor power level, including load follow modes (if employed) • reactor shutting down from power operation • shutdown in a hot standby mode • shutdown in a cold shutdown mode • shutdown in a refuelling mode or maintenance mode that opens major closures in the reactor coolant pressure boundary • shutdown in other modes or plant configurations with unique temperature, pressure or coolant inventory conditions • operation of limited duration, with some systems important to safety being unavailable For events identified by the systematic process used for this purpose, a full range of configurations and operating modes of equipment should be considered in the deterministic safety analysis. Special plant configurations may occur during major plant modifications such as plant refurbishment, lay-up, or decommissioning. These configurations should be considered, and potential events should be identified and included in the deterministic safety analysis.
Macro-Gap	SF05-02-16
Issue/Gap Description	Natural common cause events are not addressed in the Safety Report.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.2 Scope of events
Requirement Assessed	The list of events identified for the safety analysis shall include all credible: 1. component and system failures or malfunctions 2. operator errors 3. common-cause internally and externally initiated events, including those affecting multiple reactor units on a site A cut-off frequency shall be selected so that events with a frequency of occurrence less than the cut-off limit provide only a negligible contribution to the overall risk posed by the NPP. The elimination of such events from the analysis scope shall be justified and the reasons for eliminating them documented.
Macro-Gap	SF05-02-15
Issue/Gap Description	Although some PIEs listed in Table 2-1 of Part 3 of the Safety Report may be attributable to operator errors, this category of PIEs has not been explicitly identified (Gap 2).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.2 Scope of events
Requirement Assessed	The list of events identified for the safety analysis shall include all credible: 1. component and system failures or malfunctions 2. operator errors 3. common-cause internally and externally initiated events, including those affecting multiple reactor units on a site A cut-off frequency shall be selected so that events with a frequency of occurrence less than the cut-off limit provide only a negligible contribution to the overall risk posed by the NPP. The elimination of such events from the analysis scope shall be justified and the reasons for eliminating them documented.
Macro-Gap	SF05-02-15
Issue/Gap Description	1. The list of PIEs provided in Table 2-1 of Part 3 of the Safety Report covers component and system failures or malfunctions. However, the limiting case with respect to RRS working or failed has not been demonstrated for all events (e.g. Small LOCA) and therefore is considered a gap against this requirement (Gap 1)
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.2 Scope of events
Requirement Assessed	The list of events identified for the safety analysis shall include all credible: 1. component and system failures or malfunctions 2. operator errors 3. common-cause internally and externally initiated events, including those affecting multiple reactor units on a site A cut-off frequency shall be selected so that events with a frequency of occurrence less than the cut-off limit provide only a negligible contribution to the overall risk posed by the NPP. The elimination of such events from the analysis scope shall be justified and the reasons for eliminating them documented.
Macro-Gap	SF05-02-15
Issue/Gap Description	Although some common-cause internally and externally initiated events form part of the design basis for the plant, these have not been explicitly addressed in the deterministic safety analysis (Gap 3).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.2 Scope of events
Requirement Assessed	The list of events identified for the safety analysis shall include all credible: 1. component and system failures or malfunctions 2. operator errors 3. common-cause internally and externally initiated events, including those affecting multiple reactor units on a site A cut-off frequency shall be selected so that events with a frequency of occurrence less than the cut-off limit provide only a negligible contribution to the overall risk posed by the NPP. The elimination of such events from the analysis scope shall be justified and the reasons for eliminating them documented.
Macro-Gap	SF05-02-16
Issue/Gap Description	Although some PIEs listed in Table 2-1 of Part 3 of the Safety Report may be attributable to operator errors, this category of PIEs has not been explicitly identified.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.2 Scope of events
Requirement Assessed	The list of events identified for the safety analysis shall include all credible: 1. component and system failures or malfunctions 2. operator errors 3. common-cause internally and externally initiated events, including those affecting multiple reactor units on a site A cut-off frequency shall be selected so that events with a frequency of occurrence less than the cut-off limit provide only a negligible contribution to the overall risk posed by the NPP. The elimination of such events from the analysis scope shall be justified and the reasons for eliminating them documented.
Macro-Gap	SF05-02-16
Issue/Gap Description	Although some common-cause internally and externally initiated events form part of the design basis for the plant, these have not been explicitly addressed in the deterministic safety analysis.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-10-15
Issue/Gap Description	Cliff edge-effects are inherently covered in the assessment of trip coverage, however, it is not consistently addressed for quantitative acceptance criteria beyond reactor trip. The definition of design extension conditions (DECs), the classification of events that are at the border between two classes, and the scope of BDBA extending to beyond DECs are recognized in the COG guidelines for DSA [COG-09-9030]. The requirement for the analysis of DECs is introduced in REGDOC-2.5.2. Bruce A design predates this REGDOC, however some of the analyzed events considered in the design basis and included in the Safety Report, e.g. LOCA with LOECI would be classified as BDBA according to the classification scheme of REGDOC-2.4.1. DSA for BDBAs are primarily analyzed within PRA Level 2 scope to support the assessment of plant safety goals and does not normally include an assessment to search for cliff edge effects (Gap 2).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-01-15
Issue/Gap Description	At present, the deterministic safety analysis does not distinguish between these three classes of events. The focus of the Safety Report is primarily on design basis events, which include design basis accidents and AOOs. The specific event classification scheme has not been followed for deterministic safety analysis (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-02-16
Issue/Gap Description	The recommended classification for events near the border between two event classes into the higher class and consideration of the uncertainty in the event frequency in event classification is not followed.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-08-16
Issue/Gap Description	Common-cause events are not classified in the Safety Report as AOOs, DBAs or BDBAs.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-02-16
Issue/Gap Description	At present, the deterministic safety analysis does not distinguish between AOOs, DBAs and BDBAs. Therefore the event classification scheme outlined in REGDOC-2.4.1 has not been applied for deterministic safety analysis.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.2.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.2.3 Classification of events
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment), and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given provision and the existence of supporting safety analyses. Deficiencies in

	the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences
Macro-Gap	SF05-02-16
Issue/Gap Description	Cliff edge-effects are not consistently addressed for quantitative acceptance criteria beyond reactor trip. DSA for BDBAs are primarily analyzed within PRA Level 2 scope to support the assessment of plant safety goals and does not normally include an assessment to search for cliff edge effects.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-03-15
Issue/Gap Description	Gap with respect to the requirement of experimental support and demonstrating that safety margin is sufficient with accounting for

	uncertainties (See 4.3.4 compliance discussion) (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.2 Anticipated operational occurrences and design-basis accidents
Requirement Assessed	Analysis for AOOs and DBAs shall demonstrate that: 1. radiological doses to members of the public do not exceed the established limits 2. the derived acceptance criteria, established in accordance with section 4.3.4 are met Guidance The aim of safety analysis for AOOs and DBAs is to demonstrate the effectiveness of the following key safety functions: • controlling the reactor power, including shutting down the reactor and maintaining it in a shutdown state • removing heat from the core • preserving the integrity of fission product barriers • preserving component fitness for service for AOOs • ensuring that the consequences of radioactive releases are below the acceptable limits • monitoring critical safety parameters Acceptance criteria for AOOs and DBAs should include: • acceptance criteria that relate to doses to the public • derived acceptance criteria that relate to the protection of the defence-in-depth physical barriers (see section 4.3.4 and Appendix B for examples) The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, is calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose is less than or equal to one of the following dose acceptance criteria: • 0.5 millisievert for any AOO • 20 millisieverts for any DBA These dose limits apply to new NPPs. For existing reactors, the dose limits specified in the operating licences must be met. Note: New NPPs referenced in this section are effectively those first

	licensed after the issuance of RD-337, Design of New Nuclear Power Plants, in 2008. To demonstrate that the radiological consequences of an analyzed event do not exceed the limits, the doses should be calculated according to the guidance in section 4.4.4.7. Acceptance criteria for the class of events with higher frequencies of occurrence should be more stringent than those for the class of events with lower frequencies of occurrence. To demonstrate compliance with the public dose acceptance criteria for an AOO, the automatic isolation and pressure suppression functions of the containment system should not be credited, since these functions are normally considered part of Level 3 defence in depth. However, the containment passive barrier capability and normally operating containment subsystems could be credited, if they are qualified for the AOO conditions. Derived acceptance criteria have two components: qualitative and quantitative. Quantitative acceptance criteria should be developed, based on direct physical evidence and well-understood phenomena, and should account for uncertainties. Regarding the qualitative acceptance criteria (such as the examples provided in Appendix B), the following guides are applied only to AOOs: • the qualitative acceptance criteria should be satisfied without reliance on the automatic function of the safety systems, for a wide range of AOOs. The plant control systems should normally be able to correct transients and prevent damage to the plant's SSCs • the control systems should be able to maintain the plant in a stable operating state for a sufficiently long time, to allow the operator to diagnose the event, initiate required actions and, if necessary, shut the reactor down while following the applicable procedures • even though control systems may be shown to maintain the plant in a safe state following an AOO without the initiation of safety systems (Level 2 defence in depth), it should also be shown with high confidence, for all AOOs, that the safety systems can also mitigate the event without beneficial actions by the control systems (Level 3 defence in depth) Certain accidents with predicted frequency of occurrence less than 1E-5 per reactor year could be used as the design basis event for a safety system. In this case, DBA dose limits should still be met, and the analysis should also consider meeting qualitative acceptance criteria relevant to this particular safety system. The safety system performance margins should be sufficient to ensure that the DBA dose limits are met.
Macro-Gap	SF05-03-16
Issue/Gap Description	There is a lack of experimental data to support derived acceptance criteria, and it has not been demonstrated that safety margin is sufficient when

	accounting for uncertainties (see also Clause 4.3.4) (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-15
Issue/Gap Description	The acceptance criteria are not systematically supported by experimental data (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-15
Issue/Gap Description	The results of safety analysis has not been shown systematically to meet quantitative acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis (Gap 2).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-03-16
Issue/Gap Description	The acceptance criteria are not systematically supported by experimental data (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3.4 Acceptance criteria for anticipated operational occurrences and design-basis accidents
Requirement Assessed	Qualitative acceptance criteria shall be established for each AOO and DBA to confirm the effectiveness of plant systems in maintaining the integrity of physical barriers against releases of radioactive material. These qualitative acceptance criteria shall satisfy the following general principles: 1. avoid the potential for consequential failures resulting from an initiating event. 2. maintain the structures, systems and components in a configuration that permits the effective removal of residual heat. 3. prevent development of complex configurations or physical phenomena that cannot be modelled with high confidence. 4. be consistent with the design requirements for plant systems, structures and components. To demonstrate that these qualitative acceptance criteria applicable to the analyzed AOO or DBA are met, quantitative derived acceptance criteria shall be identified prior to performing the analysis. Such derived acceptance criteria shall be supported by experimental data. The results of safety analysis shall meet appropriate derived acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis. The analysis shall be performed for the event that poses the most challenges in demonstrating the meeting of derived acceptance criteria (i.e., the limiting event in an event category). Guidance In addition to the dose limits in section 4.3.2, the acceptance criteria for AOOs and DBAs also include a set of derived acceptance criteria, such as those examples of qualitative acceptance criteria identified in appendix B. These acceptance criteria are established by the designer to limit the damage to different defence barriers. Compliance with these requirements ensures that there are physical barriers preserved to limit the release of radioactive material and prevent unacceptable radiological releases following an AOO or DBA. The failure to meet a derived acceptance criterion does not necessarily mean that dose limits will be exceeded. However, if the derived acceptance criteria are met with significant margin,

	then the dose calculation can be simplified, because fission product releases are expected to be limited. The derived acceptance criteria are generally more stringent for events with a higher frequency of occurrence. For example, for most AOOs, the actions of the control systems should be able to prevent consequential degradation of any of the physical barriers to the extent that the related SSCs are no longer fit for continued service (including fuel matrix, fuel sheath/fuel cladding, reactor coolant pressure boundary or containment). More demanding requirements may be set to demonstrate the availability of a margin between the predicted value and the quantitative acceptance criteria, or to simplify an analysis (for example, to avoid having to perform complex modelling). The conditions of applicability for each additional criterion should be clearly identified. For each of the qualitative acceptance criteria, as illustrated in appendix B, quantitative acceptance criteria (or limits) should be established. These quantitative limits should: • be applicable to the particular NPP system and accident scenario • provide a clear boundary between safe states (when failure of an SSC is prevented with high confidence,) and unsafe states (when a failure of an SSC may occur) • be supported by experimental data • incorporate margins or safety factors to account for uncertainty in experimental data and relevant models When there is insufficient data to identify the transition from a safe state to an unsafe state, or to develop accurate models, then the quantitative limit for the corresponding safety requirement should be set at the boundary of the available data, provided that the established limit is conservative.
Macro-Gap	SF05-07-16
Issue/Gap Description	The results of safety analysis have not been shown systematically to meet quantitative acceptance criteria with margins sufficient to accommodate uncertainties associated with the analysis (Gap 2).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3 Acceptance criteria
Requirement Assessed	Acceptance criteria are established to serve as thresholds of safe operation in normal operation, AOOs, DBAs and, to the extent practicable, for BDBAs. The limits and conditions used by plant designers and operators should be supported by adequate experimental evidence, and be consistent with the safety analysis acceptance criteria as described in sections 4.3.1 to 4.3.4.
Macro-Gap	SF05-02-15
Issue/Gap Description	Section 1.5, Acceptance Criteria, of Part 3 of the Safety Report addresses radiological doses and derived acceptance criteria for DBAs but not explicitly for AOOs, since the limits for AOOs are currently taken to be the same as for DBAs. No reference to BDBA acceptance criteria or safety goals in the Safety Report (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.3 Acceptance criteria
Requirement Assessed	Acceptance criteria are established to serve as thresholds of safe operation in normal operation, AOOs, DBAs and, to the extent practicable, for BDBAs. The limits and conditions used by plant designers and operators should be supported by adequate experimental evidence, and be consistent with the safety analysis acceptance criteria as described in sections 4.3.1 to 4.3.4.
Macro-Gap	SF05-02-16
Issue/Gap Description	Section 1.5, Acceptance Criteria, of Part 3 of the Safety Report addresses radiological doses and derived acceptance criteria for DBAs but not explicitly for AOOs, since the limits for AOOs are currently taken to be the same as for DBAs. There is no reference to BDBA acceptance criteria or safety goals in the Safety Report.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-01-15
Issue/Gap Description	6. For some legacy analysis of small LOCA, Feedwater and Steam Supply System Failures, and Electrical System Failures not all key operating and safety system parameters are simultaneously assumed at SOE limits (Gap 4).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-01-15
Issue/Gap Description	3. The verification of the legacy analysis does not meet current standards (Gap 1). [DPT-NSAS-00013] procedure on Guidelines for Managing Reference Data Sets ensures that only verified datasets are used for deterministic safety analysis
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-01-15
Issue/Gap Description	5. Not all of the existing analyses have used validated models and computer codes that would meet the current standards (Gap 3).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-01-15
Issue/Gap Description	4. For legacy analysis of small LOCA and transition breaks analysis assumptions (such as RRS control working) should be justified (Gap 2). This practice has been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-07-16
Issue/Gap Description	Not all of the existing analyses have used validated models and computer codes that would meet the current standards.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-05-16
Issue/Gap Description	For legacy analysis of small LOCA and transition breaks, analysis assumptions (such as RRS control working), are not justified.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.1_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.1 General
Requirement Assessed	The analysis shall provide the appropriate level of confidence in demonstrating conformity with the acceptance criteria. To achieve the appropriate level of confidence, the safety analysis shall: 1. be performed by qualified analysts in accordance with an approved QA process 2. apply a systematic analysis method 3. use verified data 4. use justified assumptions 5. use verified and validated models and computer codes 6. build in a degree of conservatism 7. be subjected to a review process Guidance Section 4.4 mainly addresses analysis methods and assumptions for the deterministic safety analysis of AOOs and DBAs for Level 3 defence in depth. Similar analysis methods and assumptions can be applied for Levels 2 and 4 defence in depth (with appropriate levels of conservatism). Certain conservative rules, such as the single-failure criterion, are not applied in Level 2 and Level 4 analyses. The safety analyst has the option of selecting safety analysis methods and assumptions, as long as the regulatory requirements and expectations are satisfied. The selection of the safety analysis methods and assumptions should be such that the appropriate level of confidence can be achieved in the analysis results.
Macro-Gap	SF05-01-16
Issue/Gap Description	For some legacy analysis of small LOCA, Feedwater and Steam Supply System Failures, and Electrical System Failures not all key operating and safety system parameters are simultaneously assumed at SOE limits.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-01-15
Issue/Gap Description	The requirement of item 4 has not been applied in some of the old analyses documented in the Safety Report were produced using legacy tools predating N286.7-99 (Gap 2). New analyses follow the requirement of item 4.
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-10-15
Issue/Gap Description	Selected boundary and initial conditions for legacy analysis of SLOCA, LLOCA, breaks outside containment, electrical system failures, moderator system failures, shutdown and maintenance cooling system failures, and feedwater and steams have not been properly justified or well defined (Gap 3).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-01-15
Issue/Gap Description	7. This practice has not been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report (Gap 5).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-01-15
Issue/Gap Description	The requirements of item 3 have not been applied in Moderator System and Moderator Auxiliary System Failure legacy analysis and identification of the deuterium deflagration in moderator cover gas (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-01-15
Issue/Gap Description	6-a. The analysis of the various events include the assessment of safety margins to acceptance criteria which are selected to avoid any relevant cliff edge effects during the assessment of trip coverage. Key parameters impacting the calculated safety margins are identified and ranked for the various events in the Safety Report based on sensitivity analysis assessing the impact of a change in these parameters on the calculated safety margins. This is also recognized by the industry P&G for DSA (Section 3.8.4), For safety margins in parameters beyond trip effectiveness, cliff edge effects have not been systematically investigated (Gap 4).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-01-16
Issue/Gap Description	Selected boundary and initial conditions for legacy analysis of SLOCA, LLOCA, breaks outside containment, electrical system failures, moderator system failures, shutdown and maintenance cooling system failures, and feedwater and steams have not been properly justified or well defined.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-11-16
Issue/Gap Description	The requirement to identify the important phenomena of the analyzed event has not been applied in Moderator System and Moderator Auxiliary System Failure legacy analysis, i.e., deuterium deflagration in moderator cover gas.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-07-16
Issue/Gap Description	Accounting for uncertainties in the analysis data and models has not been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-08-16
Issue/Gap Description	The requirement for selecting computer codes, models and correlations that have been validated for the intended application has not been applied in some of the old analyses documented in the Safety Report which were produced using legacy tools predating N286.7-99.
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.2_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.2 Method for deterministic safety analysis
Requirement Assessed	The analysis method shall include the following elements: 1. identifying the scenarios to be analyzed as required to attain the analysis objectives 2. identifying the applicable acceptance criteria, safety requirements, and limits 3. identifying the important phenomena of the analyzed event 4. selecting the computational methods or computer codes, models, and correlations that have been validated for the intended applications 5. defining boundary and initial conditions 6. conducting calculations, including: a. performing sensitivity analysis and identifying, where necessary, margins to cliff-edge effects b. analyzing the event from the initial steady state up to a predefined long-term stable state, considering the guidance for duration in section 4.4.2.6 7. accounting for uncertainties in the analysis data and models 8. verifying calculation results for physical and logical consistency 9. processing and documenting the results of calculations to demonstrate conformance with the acceptance criteria Guidance The basic elements included in the safety analysis method are described in sections 4.4.2.1 to 4.4.2.9. There are three main analysis methods used in the deterministic safety analysis: • conservative analysis method, such as the method used for Level 3 defence in depth • best-estimate-plus-evaluation–of-uncertainties method, such as the method used for Level 3 defence in depth • best-estimate analysis method, such as the method used for Level 2 and Level 4 defence in depth The first and second methods above are considered as part of the

	application of conservatism in safety analysis, and are addressed in section 4.4.6. Evaluation of uncertainties is elaborated in section 4.4.2.7.
Macro-Gap	SF05-11-16
Issue/Gap Description	For safety margins in parameters beyond trip effectiveness, cliff edge effects have not been systematically investigated.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.3 Data for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis – as well as assumptions concerning the operating mode of the NPP, the availability and performance of the systems, and operator actions – shall be identified and justified. The boundary and initial conditions used as the analysis input data shall: 1. accurately reflect the NPP configuration 2. account for the effects of aging of systems, structures and components 3. account for various permissible operating modes 4. be supported by experimental data, where operational data are not available Significant uncertainties in analysis data, including those associated with NPP performance, operational measurements, and modelling parameters, shall be identified. Guidance This regulatory document requires the safety analysis be based on plant design and complete and accurate as-built information. Operational historical recorded data (such as thermal power, flow rates, temperature and pressure) should also be included, where applicable. This information should cover plant SSCs, site-specific characteristics and offsite interfaces. For an NPP in the design phase, the operational data, if needed, should be derived from generic data from operating plants of similar design, or from research or test results. For an operating NPP, the safety analysis should use plant specific operational data. The safety analysis values for each plant input parameter should be determined based on: • design specifications • tolerances • permissible ranges of variability in operation • uncertainties in measurement or evaluation for that parameter The operational data should include:

- information on component and system performance, as measured during operation or tests
- delays in control systems
- biases and drift of instrumentation
- system unavailability due to maintenance or testing

Applicable limits for NPP parameters that are used as initial and boundary conditions should be identified. The NPP parameters assumed in the safety analysis should bound the ranges of parameters allowed by the operating procedures or, in a statistical approach, cover a predetermined high percentile of each range at a predetermined high confidence level.

The following NPP parameters may be used in analysis as input data, and should be specified in the OLCs, as measured or evaluated during plant operation:

- neutronic and thermal powers, including power distribution
- pressures
- temperatures
- flows
- levels
- leakage or bypass of valves, seals, boiler tubes, and containment
- inventory of radioactive materials
- fuel sheath defects
- flux shapes
- isotopic purity of coolant and moderator (where relevant)
- neutron poison concentration
- core burnup and burnup distribution
- instrument tolerances
- instrument time constants and delays
- parameters related to SSC aging (besides accounting for aging effects on other parameters)
- position of rods, valves, dampers, doors, gates
- number of operational components, such as pumps and valves

Note: In the preparation of the data in the list above, there are some parameters (such as core burnup and burnup distribution) that are not measured directly. Core characteristics for all fuel loads should be accounted for. In this example, they are evaluated and extracted from computer simulation for which the accuracy of these tools is supported by station and experimental data. There are generally some inputs to the safety analysis that are derived or inferred from data obtained experimentally.

It should also be noted that the effects of aging include long-term mechanisms causing gradual degradation as well as mechanisms causing rapid degradation. Degradation mechanisms include thermal cycles, deformation, strain, creep, scoring, fatigue, cracking, corrosion and erosion. The allowed aging limits are part of the safety analysis input data.

	Uncertainties in plant data should be determined and recorded. These uncertainties should be considered in the uncertainty and sensitivity analyses.
Macro-Gap	SF05-01-15
Issue/Gap Description	Modeling uncertainties have not been consistently identified in Part 3 of the Safety Report (Gap 4).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.3 Data for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis – as well as assumptions concerning the operating mode of the NPP, the availability and performance of the systems, and operator actions – shall be identified and justified. The boundary and initial conditions used as the analysis input data shall: 1. accurately reflect the NPP configuration 2. account for the effects of aging of systems, structures and components 3. account for various permissible operating modes 4. be supported by experimental data, where operational data are not available Significant uncertainties in analysis data, including those associated with NPP performance, operational measurements, and modelling parameters, shall be identified. Guidance This regulatory document requires the safety analysis be based on plant design and complete and accurate as-built information. Operational historical recorded data (such as thermal power, flow rates, temperature and pressure) should also be included, where applicable. This information should cover plant SSCs, site-specific characteristics and offsite interfaces. For an NPP in the design phase, the operational data, if needed, should be derived from generic data from operating plants of similar design, or from research or test results. For an operating NPP, the safety analysis should use plant specific operational data. The safety analysis values for each plant input parameter should be determined based on: • design specifications • tolerances • permissible ranges of variability in operation • uncertainties in measurement or evaluation for that parameter The operational data should include:

	• information on component and system performance, as measured during operation or tests • delays in control systems • biases and drift of instrumentation • system unavailability due to maintenance or testing Applicable limits for NPP parameters that are used as initial and boundary conditions should be identified. The NPP parameters assumed in the safety analysis should bound the ranges of parameters allowed by the operating procedures or, in a statistical approach, cover a predetermined high percentile of each range at a predetermined high confidence level. The following NPP parameters may be used in analysis as input data, and should be specified in the OLCs, as measured or evaluated during plant operation: • neutronic and thermal powers, including power distribution • pressures • temperatures • flows • levels • leakage or bypass of valves, seals, boiler tubes, and containment • inventory of radioactive materials • fuel sheath defects • flux shapes • isotopic purity of coolant and moderator (where relevant) • neutron poison concentration • core burnup and burnup distribution • instrument tolerances • instrument time constants and delays • parameters related to SSC aging (besides accounting for aging effects on other parameters) • position of rods, valves, dampers, doors, gates • number of operational components, such as pumps and valves Note: In the preparation of the data in the list above, there are some parameters (such as core burnup and burnup distribution) that are not measured directly. Core characteristics for all fuel loads should be accounted for. In this example, they are evaluated and extracted from computer simulation for which the accuracy of these tools is supported by station and experimental data. There are generally some inputs to the safety analysis that are derived or inferred from data obtained experimentally. It should also be noted that the effects of aging include long-term mechanisms causing gradual degradation as well as mechanisms causing rapid degradation. Degradation mechanisms include thermal cycles, deformation, strain, creep, scoring, fatigue, cracking, corrosion and erosion. The allowed aging limits are part of the safety analysis input data.

	Uncertainties in plant data should be determined and recorded. These uncertainties should be considered in the uncertainty and sensitivity analyses.
Macro-Gap	SF05-01-15
Issue/Gap Description	3. This practice has been followed in most of the analyses documented in the appendices of Part 3 of the Safety Report. However, some Safety Report issues related to gaps in covering all permissible operating modes are identified (Gap 3)
Rationale	See Note 2 As part of Bruce B SF-5 review gap identified in this clause was assessed as not being a gap based on the additional information provided by Bruce Power. Bruce Power has already committed to REGDOC-2.4.1 compliance under AI 090739.

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.3 Data for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis – as well as assumptions concerning the operating mode of the NPP, the availability and performance of the systems, and operator actions – shall be identified and justified. The boundary and initial conditions used as the analysis input data shall: 1. accurately reflect the NPP configuration 2. account for the effects of aging of systems, structures and components 3. account for various permissible operating modes 4. be supported by experimental data, where operational data are not available Significant uncertainties in analysis data, including those associated with NPP performance, operational measurements, and modelling parameters, shall be identified. Guidance This regulatory document requires the safety analysis be based on plant design and complete and accurate as-built information. Operational historical recorded data (such as thermal power, flow rates, temperature and pressure) should also be included, where applicable. This information should cover plant SSCs, site-specific characteristics and offsite interfaces. For an NPP in the design phase, the operational data, if needed, should be derived from generic data from operating plants of similar design, or from research or test results. For an operating NPP, the safety analysis should use plant specific operational data. The safety analysis values for each plant input parameter should be determined based on: • design specifications • tolerances • permissible ranges of variability in operation • uncertainties in measurement or evaluation for that parameter The operational data should include:

	• information on component and system performance, as measured during operation or tests • delays in control systems • biases and drift of instrumentation • system unavailability due to maintenance or testing Applicable limits for NPP parameters that are used as initial and boundary conditions should be identified. The NPP parameters assumed in the safety analysis should bound the ranges of parameters allowed by the operating procedures or, in a statistical approach, cover a predetermined high percentile of each range at a predetermined high confidence level. The following NPP parameters may be used in analysis as input data, and should be specified in the OLCs, as measured or evaluated during plant operation: • neutronic and thermal powers, including power distribution • pressures • temperatures • flows • levels • leakage or bypass of valves, seals, boiler tubes, and containment • inventory of radioactive materials • fuel sheath defects • flux shapes • isotopic purity of coolant and moderator (where relevant) • neutron poison concentration • core burnup and burnup distribution • instrument tolerances • instrument time constants and delays • parameters related to SSC aging (besides accounting for aging effects on other parameters) • position of rods, valves, dampers, doors, gates • number of operational components, such as pumps and valves Note: In the preparation of the data in the list above, there are some parameters (such as core burnup and burnup distribution) that are not measured directly. Core characteristics for all fuel loads should be accounted for. In this example, they are evaluated and extracted from computer simulation for which the accuracy of these tools is supported by station and experimental data. There are generally some inputs to the safety analysis that are derived or inferred from data obtained experimentally. It should also be noted that the effects of aging include long-term mechanisms causing gradual degradation as well as mechanisms causing rapid degradation. Degradation mechanisms include thermal cycles, deformation, strain, creep, scoring, fatigue, cracking, corrosion and erosion. The allowed aging limits are part of the safety analysis input data.

	Uncertainties in plant data should be determined and recorded. These uncertainties should be considered in the uncertainty and sensitivity analyses.
Macro-Gap	SF05-01-15
Issue/Gap Description	1. [DPT-NSAS-00013] procedure on Guidelines for Managing Reference Data Sets ensures that only verified datasets are used for deterministic safety analysis. Some of the legacy analysis does not reflect exactly the current plant configuration (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.3_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.3 Data for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis – as well as assumptions concerning the operating mode of the NPP, the availability and performance of the systems, and operator actions – shall be identified and justified. The boundary and initial conditions used as the analysis input data shall: 1. accurately reflect the NPP configuration 2. account for the effects of aging of systems, structures and components 3. account for various permissible operating modes 4. be supported by experimental data, where operational data are not available Significant uncertainties in analysis data, including those associated with NPP performance, operational measurements, and modelling parameters, shall be identified. Guidance This regulatory document requires the safety analysis be based on plant design and complete and accurate as-built information. Operational historical recorded data (such as thermal power, flow rates, temperature and pressure) should also be included, where applicable. This information should cover plant SSCs, site-specific characteristics and offsite interfaces. For an NPP in the design phase, the operational data, if needed, should be derived from generic data from operating plants of similar design, or from research or test results. For an operating NPP, the safety analysis should use plant specific operational data. The safety analysis values for each plant input parameter should be determined based on: • design specifications • tolerances • permissible ranges of variability in operation • uncertainties in measurement or evaluation for that parameter The operational data should include:

	• information on component and system performance, as measured during operation or tests • delays in control systems • biases and drift of instrumentation • system unavailability due to maintenance or testing Applicable limits for NPP parameters that are used as initial and boundary conditions should be identified. The NPP parameters assumed in the safety analysis should bound the ranges of parameters allowed by the operating procedures or, in a statistical approach, cover a predetermined high percentile of each range at a predetermined high confidence level. The following NPP parameters may be used in analysis as input data, and should be specified in the OLCs, as measured or evaluated during plant operation: • neutronic and thermal powers, including power distribution • pressures • temperatures • flows • levels • leakage or bypass of valves, seals, boiler tubes, and containment • inventory of radioactive materials • fuel sheath defects • flux shapes • isotopic purity of coolant and moderator (where relevant) • neutron poison concentration • core burnup and burnup distribution • instrument tolerances • instrument time constants and delays • parameters related to SSC aging (besides accounting for aging effects on other parameters) • position of rods, valves, dampers, doors, gates • number of operational components, such as pumps and valves Note: In the preparation of the data in the list above, there are some parameters (such as core burnup and burnup distribution) that are not measured directly. Core characteristics for all fuel loads should be accounted for. In this example, they are evaluated and extracted from computer simulation for which the accuracy of these tools is supported by station and experimental data. There are generally some inputs to the safety analysis that are derived or inferred from data obtained experimentally. It should also be noted that the effects of aging include long-term mechanisms causing gradual degradation as well as mechanisms causing rapid degradation. Degradation mechanisms include thermal cycles, deformation, strain, creep, scoring, fatigue, cracking, corrosion and erosion. The allowed aging limits are part of the safety analysis input data.

	Uncertainties in plant data should be determined and recorded. These uncertainties should be considered in the uncertainty and sensitivity analyses.
Macro-Gap	SF05-01-15
Issue/Gap Description	2. Although ageing effects have not been comprehensively addressed in legacy analyses, newer analyses for the most impacted events account for aging effects (Gap 2).
Rationale	See Note 2 As part of Bruce B SF-5 review gap identified in this clause was assessed as not being a gap based on the additional information provided by Bruce Power. Bruce Power has already committed to REGDOC-2.4.1 compliance under AI 090739.

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.3_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.3 Data for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis – as well as assumptions concerning the operating mode of the NPP, the availability and performance of the systems, and operator actions – shall be identified and justified. The boundary and initial conditions used as the analysis input data shall: 1. accurately reflect the NPP configuration 2. account for the effects of aging of systems, structures and components 3. account for various permissible operating modes 4. be supported by experimental data, where operational data are not available Significant uncertainties in analysis data, including those associated with NPP performance, operational measurements, and modelling parameters, shall be identified. Guidance This regulatory document requires the safety analysis be based on plant design and complete and accurate as-built information. Operational historical recorded data (such as thermal power, flow rates, temperature and pressure) should also be included, where applicable. This information should cover plant SSCs, site-specific characteristics and offsite interfaces. For an NPP in the design phase, the operational data, if needed, should be derived from generic data from operating plants of similar design, or from research or test results. For an operating NPP, the safety analysis should use plant specific operational data. The safety analysis values for each plant input parameter should be determined based on: • design specifications • tolerances • permissible ranges of variability in operation • uncertainties in measurement or evaluation for that parameter The operational data should include:

- information on component and system performance, as measured during operation or tests
- delays in control systems
- biases and drift of instrumentation
- system unavailability due to maintenance or testing

Applicable limits for NPP parameters that are used as initial and boundary conditions should be identified. The NPP parameters assumed in the safety analysis should bound the ranges of parameters allowed by the operating procedures or, in a statistical approach, cover a predetermined high percentile of each range at a predetermined high confidence level.

The following NPP parameters may be used in analysis as input data, and should be specified in the OLCs, as measured or evaluated during plant operation:

- neutronic and thermal powers, including power distribution
- pressures
- temperatures
- flows
- levels
- leakage or bypass of valves, seals, boiler tubes, and containment
- inventory of radioactive materials
- fuel sheath defects
- flux shapes
- isotopic purity of coolant and moderator (where relevant)
- neutron poison concentration
- core burnup and burnup distribution
- instrument tolerances
- instrument time constants and delays
- parameters related to SSC aging (besides accounting for aging effects on other parameters)
- position of rods, valves, dampers, doors, gates
- number of operational components, such as pumps and valves

Note: In the preparation of the data in the list above, there are some parameters (such as core burnup and burnup distribution) that are not measured directly. Core characteristics for all fuel loads should be accounted for. In this example, they are evaluated and extracted from computer simulation for which the accuracy of these tools is supported by station and experimental data. There are generally some inputs to the safety analysis that are derived or inferred from data obtained experimentally.

It should also be noted that the effects of aging include long-term mechanisms causing gradual degradation as well as mechanisms causing rapid degradation. Degradation mechanisms include thermal cycles, deformation, strain, creep, scoring, fatigue, cracking, corrosion and erosion. The allowed aging limits are part of the safety analysis input data.

	Uncertainties in plant data should be determined and recorded. These uncertainties should be considered in the uncertainty and sensitivity analyses.
Macro-Gap	SF05-07-16
Issue/Gap Description	Modeling uncertainties have not been consistently identified in Part 3 of the Safety Report
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-12-15
Issue/Gap Description	DSA is usually performed until long term heat sink is established. Discussion on how and for how long a stable cold and depressurized state is maintained has not been demonstrated for the various events in the Safety Report. This should be within the scope of PRA and its supporting DSA for BDBAs (Gap 4).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-06-15
Issue/Gap Description	3. This practice has been followed in most of the analyses documented in the appendices of Part 3 of the Safety Report. However, some gaps exist regarding crediting RRS in SLOCA and transition breaks in legacy analysis and therefore this is considered a gap (Gap 3).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-05-15
Issue/Gap Description	2. For Bruce A one SAIRP issue relates to consequential failures arising during a loss of moderator inventory accident, deuterium deflagration in moderator cover gas (Gap 2).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-04-15
Issue/Gap Description	The use of more realistic assumptions for BDBAs is consistent with PRA approach and DSA for BDBAs. Some of the analyzed events in the Safety Report will be classified as BDBAs and any required revision of their analysis will adopt a more realistic analysis methodology (Gap 6).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-04-15
Issue/Gap Description	6. This practice has been followed in the analyses documented in the appendices of Part 3 of the Safety Report except for the time allowed to perform operator action for accidents involving the irradiated fuel port where operator action is credited 10 minutes after the incident. This is less than the usual 15 minutes allowed from first unambiguous indication of a problem requiring operator action from inside the main control room (Gap 5).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-04-15
Issue/Gap Description	This practice has been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report in accordance with the interpretation of the single failure criterion prevalent at the time. The analyses do not follow newer, more restrictive, interpretations of the criterion (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-11-16
Issue/Gap Description	Some gaps exist regarding crediting RRS in SLOCA and transition breaks in legacy analysis.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-04-16
Issue/Gap Description	The requirement to account for consequential failures is not fully addressed for during a loss of moderator inventory accident. Specifically, the following are not considered: (i) deuterium deflagration in moderator cover gas and (ii) impact of cobalt adjuster heatup.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.4_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.4 Assumptions for deterministic safety analysis
Requirement Assessed	Assumptions made to simplify the analysis as well as assumptions concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance 5. show that the plant can be maintained in a stable, cold and depressurized state for a prolonged period 6. credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and sufficient time to perform the required actions c. environmental conditions that do not prohibit such actions For the analysis of a BDBA, it is acceptable to use a more realistic analysis methodology consisting of assumptions that reflect the likely plant configuration, and the expected response of plant systems and operators in the analyzed accident. Guidance Assumptions are made in the input data, such as those related to the design and operating parameters as well as in the physical and numerical models implemented in the computer codes. Assumptions may be either intentionally realistic or deliberately biased in a conservative direction.

	The assumptions generally used for the Level 3 defence-in-depth analysis of AOOs and DBAs are described in sections 4.4.4.1 to 4.4.4.7. It should be noted that some of these assumptions are not necessary in the analysis of AOOs for assessing control system capability (Level 2 defence in depth,) if such an approach can be justified. For BDBA safety analysis, one objective is to demonstrate the capabilities of SSCs to meet the design requirements specified for BDBA conditions. The analysis should account for the full design capabilities of the plant, including the use of some safety and non-safety systems beyond their originally intended function (to return the potential severe accident to a controlled state, or to mitigate its consequences). The BDBA analysis assumptions on crediting and modelling plant systems and their capability during a BDBA should be consistent with the objectives of the analysis. If credit is taken for use of systems beyond their originally intended function, there should be a reasonable basis to assume they can and will be used as assumed in analysis. This basis can be obtained from the evaluation of effectiveness of these systems to operate in severe accident conditions, if they are still available.
Macro-Gap	SF05-05-16
Issue/Gap Description	The analyses do not follow newer, more restrictive, interpretations of the single failure criterion.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.5_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.5 Computer codes
Requirement Assessed	Computer codes used in the safety analysis shall be developed, validated, and used in accordance with a quality assurance program that meets the requirements of CSA N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants. Guidance The use of realistic computer codes in safety analysis is preferable, given that the use of conservative codes may produce misleading or unrealistic results. However, an extensive experimental database should be established to demonstrate the code applicability and to validate the code, thereby providing a basis for confidence in code predictions. Fully integrated models could give a more accurate representation of the event, and should be used to the extent practicable. These models address all important phenomena within a single code or code package. Sequential application of single-discipline codes is more likely to misrepresent feedback mechanisms than fully integrated models, and should be avoided unless there is a specific advantage. The selection of computer codes should consider the code applicability, the extent of code validation, and the ability to adequately represent the physical system.
Macro-Gap	SF05-01-15
Issue/Gap Description	All computer codes used in new analysis meets CSA N286.7-99. There is a gap related to the use of legacy codes and their qualifications predating N286.7-99 (Gap 1).
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.5_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.5 Computer codes
Requirement Assessed	Computer codes used in the safety analysis shall be developed, validated, and used in accordance with a quality assurance program that meets the requirements of CSA N286.7-99, Quality Assurance of Analytical, Scientific, and Design Computer Programs for Nuclear Power Plants. Guidance The use of realistic computer codes in safety analysis is preferable, given that the use of conservative codes may produce misleading or unrealistic results. However, an extensive experimental database should be established to demonstrate the code applicability and to validate the code, thereby providing a basis for confidence in code predictions. Fully integrated models could give a more accurate representation of the event, and should be used to the extent practicable. These models address all important phenomena within a single code or code package. Sequential application of single-discipline codes is more likely to misrepresent feedback mechanisms than fully integrated models, and should be avoided unless there is a specific advantage. The selection of computer codes should consider the code applicability, the extent of code validation, and the ability to adequately represent the physical system.
Macro-Gap	SF05-01-16
Issue/Gap Description	There is a gap related to the use of legacy codes and their qualifications predating N286.7-99 requirements.
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.4.6_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.4.6 Conservatism in deterministic safety analysis
Requirement Assessed	The safety analysis shall build in a degree of conservatism to off-set any uncertainties associated with both NPP initial and boundary conditions and modelling of NPP performance in the analyzed event. This conservatism shall depend on event class and shall be commensurate with the analysis objectives. Guidance Safety analysis needs to incorporate a degree of conservatism that is commensurate with the safety analysis objectives and is dependent on the event class. Conservatism in safety analysis is often necessary to cover the potential impact of uncertainties, and may be achieved through judicious application of conservative assumptions and data. The concept of conservatism is applied to Level 3 defence-in-depth safety analysis. This is to ensure that limiting assumptions are used when knowledge of the physical phenomena is insufficient. For Level 2 and Level 4 defence in depth, the safety analysis should be carried out using best-estimate assumptions, data and methods. Where this is not possible, a reasonable degree of conservatism (appropriate for the objectives of these levels) should be used, to compensate for the lack of adequate knowledge concerning the physical processes governing these events. While it is permissible – and sometimes encouraged – to use conservative codes, it is usually preferable to apply realistic (best-estimate) computer codes. Where conservative analysis results are required for Level 3 defence-in-depth (AOO and DBA) analysis, best-estimate computer codes should be used along with the assessment of modelling and input plant parameter uncertainties. The deterministic safety analysis for AOO and DBA (conservative analysis for Level 3 defence in depth) should: • apply the single-failure criterion to all safety groups, and ensure that the safety groups are environmentally and seismically qualified • use minimum allowable performance (as established in the OLCs) for safety groups • account for consequential failures that may occur as a result of the initiating event • credit the actions of process and control systems only where the systems are passive and environmentally and seismically qualified for the accident conditions

	• include the actions of process and control systems when their actions may have a detrimental effect on the consequences of the analyzed accident • credit the normally running process systems that are not affected by the analyzed accident • if operator actions are credited, demonstrate that credible “worst case” operator performance has been considered in the analysis and assessment Independent selection of all parameters at their conservative values can lead to plant states that are not physically feasible. When this could be the case, it is recommended to select conservatively those key parameters that have the strongest influence on the results in comparison with the acceptance criterion under consideration. The remaining parameters can be specified more consistently in the ensuing calculations. Each calculation should account for the impact of a particular parameter, so that the effects of all parameters can be assessed.
Macro-Gap	SF05-04-16
Issue/Gap Description	The analyses do not follow the requirements of REGDOC-2.4.1 related to the single failure criterion
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.5_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.5 Deterministic safety analysis documentation
Requirement Assessed	The safety analysis documentation shall be comprehensive and sufficiently detailed to allow for a conclusive review. The document shall include: 1. the technical basis for the analyzed event and key phenomena and processes 2. A description of the analyzed facility, including important systems and their performance as well as operator actions 3. information describing the analysis method and assumptions 4. a description of the assessments of code applicability for the analyzed event and computer code uncertainty 5. an easily understood description of the results of the analysis, and the drawing of conclusions with respect to conformance with acceptance criteria Analysis documentation shall facilitate the update of the analysis when new results become available. Guidance The review should be an independent review and conducted by suitably qualified experts. In particular, the following elements need to be included in the safety analysis documentation: • a technical basis that includes: o the objective(s) of the analysis o a description of the analyzed event, which should include a description of the NPP operating mode, action of SSCs, operator actions and significant phases of the analyzed event (note that other events bounded by the analyzed event should also be identified) o a description of safety concerns, challenges to safety, and applicable safety analysis criteria, requirements and numerical limits o identification of key phenomena significantly affected by the key parameters for the analyzed event, along with a description of the systematic process used for identification of key parameters • a description of the analyzed facility, including important systems and their performance as well as operators actions • information on the analysis method and assumptions • information demonstrating the code applicability, including (when available) evidence that codes have been validated against prototypical experiments and assessment of code accuracy as well as references to

	the relevant experimental results; demonstration that the analysis assumptions are consistent with the plant operating limits (with evidence from NPP operation and experiments demonstrating the assumed observed variances in operating parameters, and uncertainties in modelling parameters, respectively) • a description of the results of analysis, including results of sensitivity and uncertainty studies with sufficient detail to show dominant phenomena; evidence of independent verification of the inputs and the results; evidence of analysis review, including an assessment of the impact (if any) on the plant's operating limits, conditions, manuals, etc. Safety analysis documentation should be written in a manner that can be easily understood by the station staff controlling the plant's OLCs.
Macro-Gap	SF05-01-15
Issue/Gap Description	4. This practice has not been consistently followed in all the analyses documented in the appendices of Part 3 of the Safety Report (Gap 1).
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.5_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.5 Deterministic safety analysis documentation
Requirement Assessed	The safety analysis documentation shall be comprehensive and sufficiently detailed to allow for a conclusive review. The document shall include: 1. the technical basis for the analyzed event and key phenomena and processes 2. A description of the analyzed facility, including important systems and their performance as well as operator actions 3. information describing the analysis method and assumptions 4. a description of the assessments of code applicability for the analyzed event and computer code uncertainty 5. an easily understood description of the results of the analysis, and the drawing of conclusions with respect to conformance with acceptance criteria Analysis documentation shall facilitate the update of the analysis when new results become available. Guidance The review should be an independent review and conducted by suitably qualified experts. In particular, the following elements need to be included in the safety analysis documentation: • a technical basis that includes: o the objective(s) of the analysis o a description of the analyzed event, which should include a description of the NPP operating mode, action of SSCs, operator actions and significant phases of the analyzed event (note that other events bounded by the analyzed event should also be identified) o a description of safety concerns, challenges to safety, and applicable safety analysis criteria, requirements and numerical limits o identification of key phenomena significantly affected by the key parameters for the analyzed event, along with a description of the systematic process used for identification of key parameters • a description of the analyzed facility, including important systems and their performance as well as operators actions • information on the analysis method and assumptions • information demonstrating the code applicability, including (when available) evidence that codes have been validated against prototypical experiments and assessment of code accuracy as well as references to

	the relevant experimental results; demonstration that the analysis assumptions are consistent with the plant operating limits (with evidence from NPP operation and experiments demonstrating the assumed observed variances in operating parameters, and uncertainties in modelling parameters, respectively) • a description of the results of analysis, including results of sensitivity and uncertainty studies with sufficient detail to show dominant phenomena; evidence of independent verification of the inputs and the results; evidence of analysis review, including an assessment of the impact (if any) on the plant’s operating limits, conditions, manuals, etc. Safety analysis documentation should be written in a manner that can be easily understood by the station staff controlling the plant’s OLCs.
Macro-Gap	SF05-01-16
Issue/Gap Description	Descriptions of assessments of code applicability and computer code uncertainty are not documented in all the appendices of Part 3 of the Safety Report.
Rationale	See Note 2

Gap #	SF05_CNSC REGDOC 2.4.1_4.6.2_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.6.2 Update of deterministic safety analysis
Requirement Assessed	The safety analysis shall be periodically reviewed and updated to account for changes in NPP configuration, conditions (including those due to aging), operating parameters and procedures, research findings, and advances in knowledge and understanding of physical phenomena, in accordance with CNSC regulatory standard S-99, Reporting Requirements for Operating Nuclear Power Plants, or successor documents. In addition to periodic updates, the safety analysis shall also be updated following the discovery of information that may reveal a hazard that is different in nature, greater in probability, or greater in magnitude than was previously presented to the CNSC in the licensing documents. Guidance The periodic update of the safety analysis report should: • incorporate new information • address identified new issues • use current tools and methods • address the impact of modifications to the design and operating procedures that might happen over the life of the NPP Updating the safety analysis ensures that it remains valid, while taking into account: • the actual status of the NPP • permitted plant configuration and allowable operating conditions • predicted plant end-of-life state • changes to analytical methods, safety standards and knowledge that invalidate existing safety analysis In order to achieve the above objective, the following guidelines can be used in updating safety analyses: • review safety analysis methods against the applicable standards, and research findings available in Canada and internationally, to identify the elements that should be taken into account • review the changes made in the NPP data, design, operating envelope, and operating procedure, to identify the elements that need to be updated • review information on NPP commissioning and operating experience, both in Canada and worldwide, to identify relevant information that should be accounted for • review the progress in the resolution of previously identified safety

	analysis issues, to identify the impact on the safety analysis methods and results
Macro-Gap	SF05-08-15
Issue/Gap Description	Although current practices are in compliance with the requirement for review, not all analyses within Part 3 Accident Analysis [NK21-SR-01320-00003] have been fully kept up with the condition of the plant (Gap 1).
Rationale	See Note 2 As part of Bruce B SF-5 review gap identified in this clause was assessed as not being a gap based on the additional information provided by Bruce Power. Bruce Power has already committed to REGDOC-2.4.1 compliance under AI 090739.

Gap #	SF05_CNSC REGDOC 2.4.1_4.7_15
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.7 Quality of deterministic safety analysis
Requirement Assessed	Safety analysis shall be subject to a comprehensive QA program applied to all activities affecting the quality of the results. The QA program shall identify the management system or quality assurance standards to be applied and shall include documented procedures and instructions for the complete safety analysis process, including, but not limited to: 1. collection and verification of NPP data 2. verification of the computer input data 3. validation of NPP and analytical models 4. assessment of simulation results 5. documentation of analysis results Guidance All sources of data should be referenced and documented, and the various steps of the process should be recorded and archived, to allow independent checking. The safety analysis QA program should comply with regulatory requirements, codes and standards, and be consistent with the best international practices.
Macro-Gap	SF05-01-15
Issue/Gap Description	DPT-NSAS-00001] procedure on Quality Assurance of Safety Analysis establishes the quality assurance process for performing analysis work in support of nuclear safety assessment. However, use of legacy codes and their qualifications for some analysis predate N286.7 therefore do not meet the requirements for verification of computer input data nor validation of NPP and analytical models (Gap 1).
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.4.1_4.7_16
Document ID	CNSC REGDOC 2.4.1
Article/Clause	4.7 Quality of deterministic safety analysis
Requirement Assessed	Safety analysis shall be subject to a comprehensive QA program applied to all activities affecting the quality of the results. The QA program shall identify the management system or quality assurance standards to be applied and shall include documented procedures and instructions for the complete safety analysis process, including, but not limited to: 1. collection and verification of NPP data 2. verification of the computer input data 3. validation of NPP and analytical models 4. assessment of simulation results 5. documentation of analysis results Guidance All sources of data should be referenced and documented, and the various steps of the process should be recorded and archived, to allow independent checking. The safety analysis QA program should comply with regulatory requirements, codes and standards, and be consistent with the best international practices.
Macro-Gap	SF05-01-16
Issue/Gap Description	The use of legacy codes and their qualifications for some analysis predate N286.7 and therefore do not meet the current requirements for verification of computer input data nor validation of NPP and analytical models.
Rationale	See Notes 2 and 3

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.1 Dose acceptance criteria
Requirement Assessed	The acceptance criteria for normal operations are provided in section 6.4. The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, shall be calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose shall be less than or equal to the dose acceptance criteria of: 1. 0.5 millisievert (mSv) for any AOO or 2. 20 mSv for any DBA The values adopted for the dose acceptance criteria for AOOs and DBAs are consistent with accepted international practices, and take into account the recommendations of the IAEA and the International Commission on Radiological Protection.
Macro-Gap	SF05-02-15
Issue/Gap Description	Acceptance Criteria, of Part 3 of the Safety Report addresses radiological doses and derived acceptance criteria for DBAs but not explicitly for AOOs, since the limits for AOOs are currently taken to be the same as for DBAs (Gap 1).
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.1 Dose acceptance criteria
Requirement Assessed	The acceptance criteria for normal operations are provided in section 6.4. The committed whole-body dose for average members of the critical groups who are most at risk, at or beyond the site boundary, shall be calculated in the deterministic safety analysis for a period of 30 days after the analyzed event. This dose shall be less than or equal to the dose acceptance criteria of: 1. 0.5 millisievert (mSv) for any AOO or 2. 20 mSv for any DBA The values adopted for the dose acceptance criteria for AOOs and DBAs are consistent with accepted international practices, and take into account the recommendations of the IAEA and the International Commission on Radiological Protection.
Macro-Gap	SF05-02-16
Issue/Gap Description	Acceptance Criteria, of Part 3 of the Safety Report addresses radiological doses and derived acceptance criteria for DBAs but not explicitly for AOOs, since the limits for AOOs are currently taken to be the same as for DBAs.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.3_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF05-02-15
Issue/Gap Description	DSA in the Safety Report does not distinguish between AOO and DBA and does not address BDBAs explicitly (Gap 1). DECs were not considered in the design basis; however, the design basis includes some event sequences that would be categorized as BDBAs.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_4.2.3_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	4.2.3 Safety analyses
Requirement Assessed	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.
Macro-Gap	SF05-02-16
Issue/Gap Description	Deterministic Safety Analysis in the Safety Report does not distinguish between AOOs and DBAs and does not address BDBAs explicitly.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.1 Application of defence in depth
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

	and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures. Level Five The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response. Guidance IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth. Guidance on performing a systematic assessment of the defence in depth can be obtained from the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants. The application of defence in depth in the design should ensure the following: • The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety. • The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs). • The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment) and adequate justification should exist for such design choices. • The design (e.g., in safety design guides, management system programs) should provide: • levels of defence in depth that are addressed by individual SSCs • supporting analysis and calculation • evaluation of operating procedures • The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity. • The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design. • Special attention should be given to the feasibility of a given

	provision and the existence of supporting safety analyses. Deficiencies in the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences of the accident. The independence between all levels of defence should be achieved, in particular, through diverse provisions. The strengthening of each of these levels separately would provide, as far as reasonably achievable, an overall reinforcement of defence in depth. For example, the use of dedicated systems to deal with DECs ensures the independence of the fourth defence level.
Macro-Gap	SF05-02-15
Issue/Gap Description	Level 2 defence-in-depth is not demonstrated explicitly for AOOs (Gap 1).
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.1 Application of defence in depth
Requirement Assessed	The design of an NPP shall incorporate defence in depth. The levels of defence in depth shall be independent to the extent practicable. Defence in depth shall be achieved at the design phase through the application of design provisions specific to the five levels of defence. Level One Achievement of Level one defence in depth shall include conservative design and high-quality construction to provide confidence that plant failures and deviations from normal operations are minimized and accidents are prevented. This shall entail careful attention to selection of appropriate design codes and materials, design procedures, equipment qualification, control of component fabrication and plant construction, and use of operational experience. Level Two Level two shall be achieved by controlling plant behaviour during and following a postulated initiating event (PIE) using both inherent and engineered design features to minimize or exclude uncontrolled transients to the extent possible. Level Three Achievement of Level three defence in depth shall include the provision of inherent safety features, fail-safe design, engineered design features, and procedures that minimize the consequences of DBAs. These provisions shall be capable of leading the plant first to a controlled state, and then to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material. Automatic activation of the engineered design features shall minimize the need for operator actions in the early phase of a DBA. Level Four Level four shall be achieved by providing equipment and procedures to manage accidents and mitigate their consequences as far as practicable. Most importantly, adequate protection shall be provided for the confinement function by way of a robust containment design. This includes the use of complementary design features to prevent accident progression

and to mitigate the consequences of DECs. The confinement function shall be further protected by severe accident management procedures.

Level Five

The design shall provide adequately equipped emergency support facilities, and plans for onsite and offsite emergency response.

Guidance

IAEA INSAG-10, Defence in Depth in Nuclear Safety, provides information regarding the concept and application of defence in depth.

Guidance on performing a systematic assessment of the defence in depth can be obtained from
the IAEA safety reports series No. 46, Assessment of Defence in Depth for Nuclear Power Plants.

The application of defence in depth in the design should ensure the following:

- The approach to defence in depth used in the design should ensure that all aspects of design at the SSCs level have been covered, with emphasis on SSCs that are important to safety.
- The defence in depth should not be significantly degraded if the SSC has multiple functions (e.g., for CANDU reactors, the moderator and end-shield cooling systems may serve the functions of a process system and include the functions of mitigating DECs).
- The principle of multiple physical barriers to the release of radioactive material should be incorporated in the design; there should be a limited number of cases where there is a reduction in the number of physical barriers (as may be the case where some components carrying radioactive material serve the function of primary coolant barrier and containment) and adequate justification should exist for such design choices.
- The design (e.g., in safety design guides, management system programs) should provide:
- levels of defence in depth that are addressed by individual SSCs
- supporting analysis and calculation
- evaluation of operating procedures
- The safety analysis should demonstrate that the challenges to the physical barriers do not exceed their physical capacity.
- The structure for defence in depth provisions at each level of defence should be established for a given plant design, and the evaluation of the design from the point of view of maintaining each safety function should be carried out. This evaluation should consider each and every one of the provisions for mitigation of a given challenge mechanism, and confirm that it is well founded, sufficient, feasible, and correctly engineered within the design.
- Special attention should be given to the feasibility of a given

	provision and the existence of supporting safety analyses. Deficiencies in the completeness of the supporting safety analyses should be documented and flagged as issues to be queried. To ensure that different levels of defence are independently effective, any design features that aim to prevent an accident should not belong to the same level of defence as design features that aim to mitigate the consequences of the accident. The independence between all levels of defence should be achieved, in particular, through diverse provisions. The strengthening of each of these levels separately would provide, as far as reasonably achievable, an overall reinforcement of defence in depth. For example, the use of dedicated systems to deal with DECs ensures the independence of the fourth defence level.
Macro-Gap	SF05-02-16
Issue/Gap Description	Level 2 defence-in-depth is not demonstrated explicitly for AOOs.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF05-02-15
Issue/Gap Description	DSA in the Safety Report does not distinguish between AOO and DBA and does not address BDBAs explicitly (Gap 1).

Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF05-02-15
Issue/Gap Description	Bruce A design basis includes some event sequences that would be categorized as BDBAs (e.g. Large LOCA with LOECI), however, Bruce A

	does not meet this requirement intended for new builds (Gap 2).
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.4 Radiation protection and acceptance criteria
Requirement Assessed	Achievement of the general nuclear safety objective (discussed in section 4.1) depends on all actual and potential sources of radiation being identified, and on provision being made to ensure that sources are kept under strict technical and administrative control. Radiation doses to the public and to site personnel shall be as low as reasonably achievable. During normal operation, including maintenance and decommissioning, doses shall be regulated by the limits prescribed in the Radiation Protection Regulations. The design shall include provisions for the prevention and mitigation of radiation exposures resulting from DBAs and DECs. The design shall also ensure that potential radiation doses to the public from AOOs and DBAs do not exceed dose acceptance criteria provided in section 4.2.1. The calculated overall risk to the public shall meet the safety goals in section 4.2.2. Guidance A detailed radiation dose assessment should include estimated annual collective and individual effective and equivalent radiation doses to site personnel and members of the public for normal operation, potential radiation doses to the public for AOOs and DBAs, and potential releases into the environment for DECs. The assessment process should be clearly documented and should include the process for consideration and evaluation of dose-reduction changes in the NPP design. Radiation doses resulting from the operation of the NPP should be reduced by means of engineered controls and radiation protection measures to levels such that any further expenditure on design, construction and operational measures would not be warranted by the expected reduction in radiation doses. The radiation dose assessment should include the expected occupancy of the NPP's radiation areas, along with estimated annual person-Sievert doses associated with major functions, including radioactive waste handling, normal maintenance, special maintenance, refuelling and in-service inspection. Such assessments should include information as to how ALARA and operating experience are used in the design to deal with dose-significant contributors.
Macro-Gap	SF05-02-16
Issue/Gap Description	DSA in the Safety Report does not distinguish between AOO and DBA and does not address BDBAs explicitly.

Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.6.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.6.1 Requirements for multiple units
Requirement Assessed	The design shall take due account of challenges to multiple units at a site. Specifically, the risk associated with common-cause events affecting more than one unit at a time shall be considered. Guidance The presence of multiple units at a site, or common-cause events could exacerbate challenges that the plant personnel would face during an accident. The events and consequences of an accident at one unit may affect the accident progression or hamper accident management activities at the neighbouring unit; available resources (personnel, equipment and consumable resources) would need to be shared among several units. These challenges should be identified and the available resources and mitigation strategies shown to be adequate.
Macro-Gap	SF05-02-15
Issue/Gap Description	Common-cause events are not analyzed explicitly in Part 3 of the Safety Report (Gap 1). This gap is scheduled to be considered early within Safety Report update towards the compliance with REGDOC-2.4.1.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_6.6.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	6.6.1 Requirements for multiple units
Requirement Assessed	The design shall take due account of challenges to multiple units at a site. Specifically, the risk associated with common-cause events affecting more than one unit at a time shall be considered. Guidance The presence of multiple units at a site, or common-cause events could exacerbate challenges that the plant personnel would face during an accident. The events and consequences of an accident at one unit may affect the accident progression or hamper accident management activities at the neighbouring unit; available resources (personnel, equipment and consumable resources) would need to be shared among several units. These challenges should be identified and the available resources and mitigation strategies shown to be adequate.
Macro-Gap	SF05-02-16
Issue/Gap Description	Common-cause events are not analyzed explicitly in Part 3 of the Safety Report.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_7.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4 Postulated initiating events
Requirement Assessed	The design for the NPP shall apply a systematic approach to identifying a comprehensive set of postulated initiating events, such that all foreseeable events with the potential for serious consequences or with a significant frequency of occurrence are anticipated and considered. Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs as well as operator errors, common-cause internal hazards, and external hazards. For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site. Guidance The postulated initiating events (PIEs) are identified using engineering judgment and deterministic and probabilistic assessment. A justification of the extent of usage of deterministic safety analyses and probabilistic safety analyses should be provided, in order to show that all foreseeable events have been considered. Sufficient information should be provided regarding the methods used to identify PIEs, their scope and classification. In cases where the identification methods have made use of analytical tools (e.g., master logic diagrams, hazard and operability analysis, failure modes and effect analysis), detailed information is expected to be presented. A systematic approach to event classification should consider all internal and external events, all normal operating configurations, various plant and site conditions, and failure in other plant systems (e.g., storage for irradiated fuel, and tanks for radioactive substances). The design should take into account failure of equipment that is not part of the NPP, if the failure has a significant impact on nuclear safety. CNSC REGDOC-2.4.1, Deterministic Safety Analysis and REGDOC-2.4.2, Probabilistic Safety Assessments, provide the requirements and guidance for establishing the scope of PIEs, and for classifying the PIEs in accordance with their anticipated frequencies, and other factors, as appropriate. For further information on the safety analysis for the identified PIEs, refer to section 9.0 of this document.
Macro-Gap	SF05-02-15

Issue/Gap Description	A systematic event identification process is not well documented and/or demonstrated. Postulated initiating events are not categorized into AOOs, DBAs or BDBAs (Gap 1). For more details, see Assessment against REGDOC-2.4.1
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_7.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4 Postulated initiating events
Requirement Assessed	The design for the NPP shall apply a systematic approach to identifying a comprehensive set of postulated initiating events, such that all foreseeable events with the potential for serious consequences or with a significant frequency of occurrence are anticipated and considered. Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs as well as operator errors, common-cause internal hazards, and external hazards. For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site. Guidance The postulated initiating events (PIEs) are identified using engineering judgment and deterministic and probabilistic assessment. A justification of the extent of usage of deterministic safety analyses and probabilistic safety analyses should be provided, in order to show that all foreseeable events have been considered. Sufficient information should be provided regarding the methods used to identify PIEs, their scope and classification. In cases where the identification methods have made use of analytical tools (e.g., master logic diagrams, hazard and operability analysis, failure modes and effect analysis), detailed information is expected to be presented. A systematic approach to event classification should consider all internal and external events, all normal operating configurations, various plant and site conditions, and failure in other plant systems (e.g., storage for irradiated fuel, and tanks for radioactive substances). The design should take into account failure of equipment that is not part of the NPP, if the failure has a significant impact on nuclear safety. CNSC REGDOC-2.4.1, Deterministic Safety Analysis and REGDOC-2.4.2, Probabilistic Safety Assessments, provide the requirements and guidance for establishing the scope of PIEs, and for classifying the PIEs in accordance with their anticipated frequencies, and other factors, as appropriate. For further information on the safety analysis for the identified PIEs, refer to section 9.0 of this document.
Macro-Gap	SF05-02-16

Issue/Gap Description	Common-cause events are not analyzed explicitly in Part 3 of the Safety Report.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_7.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.4 Postulated initiating events
Requirement Assessed	The design for the NPP shall apply a systematic approach to identifying a comprehensive set of postulated initiating events, such that all foreseeable events with the potential for serious consequences or with a significant frequency of occurrence are anticipated and considered. Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs as well as operator errors, common-cause internal hazards, and external hazards. For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site. Guidance The postulated initiating events (PIEs) are identified using engineering judgment and deterministic and probabilistic assessment. A justification of the extent of usage of deterministic safety analyses and probabilistic safety analyses should be provided, in order to show that all foreseeable events have been considered. Sufficient information should be provided regarding the methods used to identify PIEs, their scope and classification. In cases where the identification methods have made use of analytical tools (e.g., master logic diagrams, hazard and operability analysis, failure modes and effect analysis), detailed information is expected to be presented. A systematic approach to event classification should consider all internal and external events, all normal operating configurations, various plant and site conditions, and failure in other plant systems (e.g., storage for irradiated fuel, and tanks for radioactive substances). The design should take into account failure of equipment that is not part of the NPP, if the failure has a significant impact on nuclear safety. CNSC REGDOC-2.4.1, Deterministic Safety Analysis and REGDOC-2.4.2, Probabilistic Safety Assessments, provide the requirements and guidance for establishing the scope of PIEs, and for classifying the PIEs in accordance with their anticipated frequencies, and other factors, as appropriate. For further information on the safety analysis for the identified PIEs, refer to section 9.0 of this document.
Macro-Gap	SF05-02-16

Issue/Gap Description	A systematic event identification process is not well documented and/or demonstrated. Postulated initiating events are not categorized into AOOs, DBAs or BDBAs.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_7.6.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.2 Single-failure criterion
Requirement Assessed	All safety groups shall function in the presence of a single failure. The single-failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure as well as: 1. all failures caused by that single failure 2. all identifiable but non-detectable failures, including those in the non-tested components 3. all failures and spurious system actions that cause (or are caused by) the PIE Each safety group shall be able to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. Analysis of all possible single failures, and all associated consequential failures, shall be conducted for each component of each safety group until all safety groups have been considered. Unintended actions and failure of passive components shall be considered as two of the modes of failure of a safety group. The single failure shall be assumed to occur prior to the PIE, or at any time during the mission time for which the safety group is required to function following the PIE. Passive components may be exempt from this requirement. Exceptions to the single-failure criterion shall be infrequent, and clearly justified. Exemptions for passive components may be applied only to those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by the PIE. Design documentation shall include justification of such exemptions, by analysis, testing or a combination of analysis and testing. The justification shall take loads and environmental conditions into account as well as the total period of time after the PIE for which the functioning of the component is necessary. Check valves shall be considered to be active components if they must change state following a PIE.

	Guidance The application of the single-failure criterion (SFC) in design should follow a systematic approach applied to all safety groups. The approach should be adequately verified, such as by using failure modes and effects analysis. The SSCs inside the safety group should include both the primary SSCs and the supporting SSCs. The detectability of failures is implicit in the application of the SFC. Detectability is a function of the system design and the specified tests. A failure that cannot be detected through periodic testing, or revealed by alarm or anomalous indication, is non-detectable. An objective in a single-failure analysis is to identify non-detectable failures. To deal with identifiable but non-detectable failures, the following actions should be considered: • preferred action: the system or the test scheme should be redesigned to make the failure detectable • alternative action: when analyzing the effect of each single failure, all identified non- detectable failures should be assumed to have occurred. Therefore, the design should take appropriate measures to address these non-detectable failures, such as adequate redundancy and diversity Justification in support of an exception to the SFC should consider the consequences of failure, practicality of alternatives, added complexity and operational considerations. The integrated effect of all exceptions should not significantly degrade safety; in particular, defence in depth should be preserved. For passive components that are exempt from the SFC, the following should be considered in order to demonstrate a high degree of performance assurance: • adequate testing during the manufacturing stage • sample testing from those components received from the manufacturer • adequate testing during construction and commissioning stages • necessary testing to verify their reliability after the components have been removed from service during the operation stage Any consideration for an exception to the SFC during testing and maintenance should fall into one of the following permissible categories: • the safety function is provided by two redundant, independent systems (e.g., two redundant, fully effective, independent cooling means) • the expected duration of testing and maintenance is shorter than the time available before the function is required following an initiating event (e.g., spent fuel storage pool cooling) • the loss of safety function is partial and unlikely to lead to

	significant increase in risk even in the event of failure (e.g., small area containment isolation) • the loss of system redundancy has minor safety significance (e.g., control room air filtering) • the loss of system redundancy may slightly increase PIE frequency, but does not impact accident progression (e.g., leak detection) A request for an exception during testing and maintenance should also be supported by a satisfactory reliability argument covering the allowable outage time. The OLCs should clearly state the allowable testing and maintenance time, along with any additional operational restrictions, such as suspension of additional testing or maintenance on a backup system for the duration of the exception.
Macro-Gap	SF05-04-15
Issue/Gap Description	The analyses do not follow newer, more restrictive, interpretations of the single failure criterion (Gap 1).
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_7.6.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	7.6.2 Single-failure criterion
Requirement Assessed	All safety groups shall function in the presence of a single failure. The single-failure criterion requires that each safety group can perform all safety functions required for a PIE in the presence of any single component failure as well as: 1. all failures caused by that single failure 2. all identifiable but non-detectable failures, including those in the non-tested components 3. all failures and spurious system actions that cause (or are caused by) the PIE Each safety group shall be able to perform the required safety functions under the worst permissible systems configuration, taking into account such considerations as maintenance, testing, inspection and repair, and equipment outage. Analysis of all possible single failures, and all associated consequential failures, shall be conducted for each component of each safety group until all safety groups have been considered. Unintended actions and failure of passive components shall be considered as two of the modes of failure of a safety group. The single failure shall be assumed to occur prior to the PIE, or at any time during the mission time for which the safety group is required to function following the PIE. Passive components may be exempt from this requirement. Exceptions to the single-failure criterion shall be infrequent, and clearly justified. Exemptions for passive components may be applied only to those components that are designed and manufactured to high standards of quality, that are adequately inspected and maintained in service, and that remain unaffected by the PIE. Design documentation shall include justification of such exemptions, by analysis, testing or a combination of analysis and testing. The justification shall take loads and environmental conditions into account as well as the total period of time after the PIE for which the functioning of the component is necessary. Check valves shall be considered to be active components if they must change state following a PIE.

	Guidance The application of the single-failure criterion (SFC) in design should follow a systematic approach applied to all safety groups. The approach should be adequately verified, such as by using failure modes and effects analysis. The SSCs inside the safety group should include both the primary SSCs and the supporting SSCs. The detectability of failures is implicit in the application of the SFC. Detectability is a function of the system design and the specified tests. A failure that cannot be detected through periodic testing, or revealed by alarm or anomalous indication, is non-detectable. An objective in a single-failure analysis is to identify non-detectable failures. To deal with identifiable but non-detectable failures, the following actions should be considered: • preferred action: the system or the test scheme should be redesigned to make the failure detectable • alternative action: when analyzing the effect of each single failure, all identified non- detectable failures should be assumed to have occurred. Therefore, the design should take appropriate measures to address these non-detectable failures, such as adequate redundancy and diversity Justification in support of an exception to the SFC should consider the consequences of failure, practicality of alternatives, added complexity and operational considerations. The integrated effect of all exceptions should not significantly degrade safety; in particular, defence in depth should be preserved. For passive components that are exempt from the SFC, the following should be considered in order to demonstrate a high degree of performance assurance: • adequate testing during the manufacturing stage • sample testing from those components received from the manufacturer • adequate testing during construction and commissioning stages • necessary testing to verify their reliability after the components have been removed from service during the operation stage Any consideration for an exception to the SFC during testing and maintenance should fall into one of the following permissible categories: • the safety function is provided by two redundant, independent systems (e.g., two redundant, fully effective, independent cooling means) • the expected duration of testing and maintenance is shorter than the time available before the function is required following an initiating event (e.g., spent fuel storage pool cooling) • the loss of safety function is partial and unlikely to lead to

	significant increase in risk even in the event of failure (e.g., small area containment isolation) • the loss of system redundancy has minor safety significance (e.g., control room air filtering) • the loss of system redundancy may slightly increase PIE frequency, but does not impact accident progression (e.g., leak detection) A request for an exception during testing and maintenance should also be supported by a satisfactory reliability argument covering the allowable outage time. The OLCs should clearly state the allowable testing and maintenance time, along with any additional operational restrictions, such as suspension of additional testing or maintenance on a backup system for the duration of the exception.
Macro-Gap	SF05-04-16
Issue/Gap Description	The analyses do not follow newer, more restrictive, interpretations of the single failure criterion.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_8.4.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.4.1 Reactor trip parameters
Requirement Assessed	The design authority shall specify derived acceptance criteria for reactor trip parameter effectiveness for all AOOs and DBAs, and shall perform a safety analysis to demonstrate the effectiveness of the means of shutdown. For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all AOOs and DBAs in time to meet the respective derived acceptance criteria. Where a direct trip parameter does not exist for a given credited means, there shall be two diverse trip parameters specified for that means. For all AOOs and DBAs, there shall be at least two diverse trip parameters unless it can be shown that failure to trip will not lead to unacceptable consequences. There shall be no gap in trip coverage within the OLCs for any operating condition (such as power, temperature), taking into account plant aging. This shall be ensured by the provision of additional trip parameters if necessary. A different level of effectiveness may be acceptable for the additional trip parameters. The extent of trip coverage provided by all available parameters shall be documented for the entire spectrum of failures for each set of PIEs. An assessment of the accuracy and the potential failure modes of the trip parameters shall be provided in the design documentation. Guidance The effectiveness of trip parameters should be assessed through safety analysis performed in accordance with REGDOC-2.4.1, Deterministic Safety Analysis. Trip coverage should be demonstrated across the full range of operating states, for all credited shutdown means and all credited trip parameters. Note that the number of credited shutdown means and the number of credited trip parameters can vary with the event, the reactor design, and whether there is a direct trip available. Defining derived acceptance criteria appropriate to a particular design is the responsibility of the design authority. CNSC REGDOC-2.4.1, Deterministic Safety Analysis, provides the requirements. Derived acceptance criteria should be defined separately for AOOs and

	DBAs. The derived acceptance criteria should be set to give an appropriate level of confidence that a fundamental safety function is assured, or that a barrier to fission product release will not fail. The derived acceptance criteria should: • be quantifiable and well understood • account for the fact that the safety analysis is stylized, and the plant condition at the time of the accident may be significantly different from the analyzed state • cover uncertainties in analysis, input plant and analysis parameters as well as code validation Direct trips are the preferred means of actuating a shutdown means, due to their robustness and low dependence on calculational models. Diverse trip parameters measure different physical variables on the reactor, thus providing additional protection against common mode failure. Where it is impracticable to provide full diversity of trip parameters, different measurement locations, different instrument types and different processing computers should be provided. Manual trip is considered an acceptable trip parameter, if the operator has adequate time to initiate the shutdown action following unambiguous indication of the need to perform the action (in accordance with section 8.10.4). It is the responsibility of the design authority to identify and justify those trip parameters that can be considered “direct”. The design authority should also demonstrate that any trip parameters that are a measure of the event, but not a measure of the challenge to acceptance criteria, cannot be “masked” or “blinded” by control system action or other means. Trips that are dependent on a number of measured variables, such as low DNBR (departure from nucleate boiling ratio) trips in PWRs can only be considered direct if all the variables are direct. Guidance on applying the requirements for number and diversity of trip parameters is given in REGDOC-2.4.1, Deterministic Safety Analysis. REGDOC-2.4.1 also provides the minimum expectations for the number of trip parameters. A manual reactor trip can be considered to be equivalent to a trip parameter, if the requirements for crediting operator action from the main control room are met (see section 8.10.4) and the reliability of manual shutdown meets the reliability requirements for an automatic trip.
Macro-Gap	SF05-03-15
Issue/Gap Description	Acceptance criteria are not explicitly specified for AOOs (Gap 1). See assessment against REGDOC-2.4.1 requirements.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_8.4.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.4.1 Reactor trip parameters
Requirement Assessed	The design authority shall specify derived acceptance criteria for reactor trip parameter effectiveness for all AOOs and DBAs, and shall perform a safety analysis to demonstrate the effectiveness of the means of shutdown. For each credited means of shutdown, the design shall specify a direct trip parameter to initiate reactor shutdown for all AOOs and DBAs in time to meet the respective derived acceptance criteria. Where a direct trip parameter does not exist for a given credited means, there shall be two diverse trip parameters specified for that means. For all AOOs and DBAs, there shall be at least two diverse trip parameters unless it can be shown that failure to trip will not lead to unacceptable consequences. There shall be no gap in trip coverage within the OLCs for any operating condition (such as power, temperature), taking into account plant aging. This shall be ensured by the provision of additional trip parameters if necessary. A different level of effectiveness may be acceptable for the additional trip parameters. The extent of trip coverage provided by all available parameters shall be documented for the entire spectrum of failures for each set of PIEs. An assessment of the accuracy and the potential failure modes of the trip parameters shall be provided in the design documentation. Guidance The effectiveness of trip parameters should be assessed through safety analysis performed in accordance with REGDOC-2.4.1, Deterministic Safety Analysis. Trip coverage should be demonstrated across the full range of operating states, for all credited shutdown means and all credited trip parameters. Note that the number of credited shutdown means and the number of credited trip parameters can vary with the event, the reactor design, and whether there is a direct trip available. Defining derived acceptance criteria appropriate to a particular design is the responsibility of the design authority. CNSC REGDOC-2.4.1, Deterministic Safety Analysis, provides the requirements. Derived acceptance criteria should be defined separately for AOOs and

	DBAs. The derived acceptance criteria should be set to give an appropriate level of confidence that a fundamental safety function is assured, or that a barrier to fission product release will not fail. The derived acceptance criteria should: • be quantifiable and well understood • account for the fact that the safety analysis is stylized, and the plant condition at the time of the accident may be significantly different from the analyzed state • cover uncertainties in analysis, input plant and analysis parameters as well as code validation Direct trips are the preferred means of actuating a shutdown means, due to their robustness and low dependence on calculational models. Diverse trip parameters measure different physical variables on the reactor, thus providing additional protection against common mode failure. Where it is impracticable to provide full diversity of trip parameters, different measurement locations, different instrument types and different processing computers should be provided. Manual trip is considered an acceptable trip parameter, if the operator has adequate time to initiate the shutdown action following unambiguous indication of the need to perform the action (in accordance with section 8.10.4). It is the responsibility of the design authority to identify and justify those trip parameters that can be considered “direct”. The design authority should also demonstrate that any trip parameters that are a measure of the event, but not a measure of the challenge to acceptance criteria, cannot be “masked” or “blinded” by control system action or other means. Trips that are dependent on a number of measured variables, such as low DNBR (departure from nucleate boiling ratio) trips in PWRs can only be considered direct if all the variables are direct. Guidance on applying the requirements for number and diversity of trip parameters is given in REGDOC-2.4.1, Deterministic Safety Analysis. REGDOC-2.4.1 also provides the minimum expectations for the number of trip parameters. A manual reactor trip can be considered to be equivalent to a trip parameter, if the requirements for crediting operator action from the main control room are met (see section 8.10.4) and the reliability of manual shutdown meets the reliability requirements for an automatic trip.
Macro-Gap	SF05-03-16
Issue/Gap Description	Acceptance criteria are not explicitly specified for AOOs.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.1 General
Requirement Assessed	A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals. Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included. The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.
Macro-Gap	SF05-02-15
Issue/Gap Description	A systematic methodology for event identification is not demonstrated.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.1 General
Requirement Assessed	A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals. Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included. The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.
Macro-Gap	SF05-02-16
Issue/Gap Description	A systematic methodology for event identification is not demonstrated.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.2_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.2 Analysis objectives
Requirement Assessed	The safety analysis shall be iterative with the design process, and result in two reports: a preliminary safety analysis report, and a final safety analysis report. The preliminary safety analysis shall assist in the establishment of the design-basis requirements for the items important to safety, and demonstrate whether the plant design meets applicable requirements. The final safety analysis shall: 1. reflect the as-built plant 2. account for postulated aging effects on SSCs important to safety 3. demonstrate that the design can withstand and effectively respond to identified PIEs 4. demonstrate the effectiveness of the safety systems and safety support systems 5. derive the OLCs for the plant, including: a. operational limits and set points important to safety b. allowable operating configurations, and constraints for operational procedures 6. establish requirements for emergency response and accident management 7. determine post-accident environmental conditions, including radiation fields and worker doses, to confirm that operators are able to carry out the actions credited in the analysis 8. demonstrate that the design incorporates sufficient safety margins 9. confirm that the dose and derived acceptance criteria are met for all AOOs and DBAs 10. demonstrate that all safety goals have been met Guidance The Class I Nuclear Facilities Regulations requires a preliminary safety analysis report demonstrating the adequacy of the NPP design to be submitted in support of an application for a licence to construct a Class I nuclear facility. A final safety analysis report demonstrating the adequacy of the design is required for an application for a licence to operate a Class I nuclear facility.
Macro-Gap	SF05-02-15
Issue/Gap Description	The dose and other acceptance criteria for AOOs are not explicitly assessed in Part 3 of the Safety Report (Gap 1).
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.2_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.2 Analysis objectives
Requirement Assessed	The safety analysis shall be iterative with the design process, and result in two reports: a preliminary safety analysis report, and a final safety analysis report. The preliminary safety analysis shall assist in the establishment of the design-basis requirements for the items important to safety, and demonstrate whether the plant design meets applicable requirements. The final safety analysis shall: 1. reflect the as-built plant 2. account for postulated aging effects on SSCs important to safety 3. demonstrate that the design can withstand and effectively respond to identified PIEs 4. demonstrate the effectiveness of the safety systems and safety support systems 5. derive the OLCs for the plant, including: a. operational limits and set points important to safety b. allowable operating configurations, and constraints for operational procedures 6. establish requirements for emergency response and accident management 7. determine post-accident environmental conditions, including radiation fields and worker doses, to confirm that operators are able to carry out the actions credited in the analysis 8. demonstrate that the design incorporates sufficient safety margins 9. confirm that the dose and derived acceptance criteria are met for all AOOs and DBAs 10. demonstrate that all safety goals have been met Guidance The Class I Nuclear Facilities Regulations requires a preliminary safety analysis report demonstrating the adequacy of the NPP design to be submitted in support of an application for a licence to construct a Class I nuclear facility. A final safety analysis report demonstrating the adequacy of the design is required for an application for a licence to operate a Class I nuclear facility.
Macro-Gap	SF05-02-16
Issue/Gap Description	The dose and other acceptance criteria for AOOs are not explicitly assessed in Part 3 of the Safety Report.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.4 Deterministic safety analysis
Requirement Assessed	The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis.
Macro-Gap	SF05-02-15
Issue/Gap Description	Assessment against REGDOC-2.4.1 identified gaps in the deterministic safety analysis that are related to; • event identification and classification (Gap 1),
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.4 Deterministic safety analysis
Requirement Assessed	The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis.
Macro-Gap	SF05-03-15
Issue/Gap Description	Assessment against REGDOC-2.4.1 identified gaps in the deterministic safety analysis that are related to treatment of modeling uncertainty (Gap 2)
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.4_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.4 Deterministic safety analysis
Requirement Assessed	The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis.
Macro-Gap	SF05-01-15
Issue/Gap Description	Assessment against REGDOC-2.4.1 identified gaps in the deterministic safety analysis that are related to; • the use of legacy tools for some analysis (Gap 3). ………………
Rationale	See Notes 1, 2 and 3

Gap #	SF05_CNSC REGDOC 2.5.2_9.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.4 Deterministic safety analysis
Requirement Assessed	The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis.
Macro-Gap	SF05-07-16
Issue/Gap Description	Assessment against REGDOC-2.4.1 identified gaps in the deterministic safety analysis that are related to treatment of modeling uncertainty.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.4 Deterministic safety analysis
Requirement Assessed	The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis.
Macro-Gap	SF05-02-16
Issue/Gap Description	Assessment against REGDOC-2.4.1 identified gaps in the deterministic safety analysis that are related to event identification and classification.
Rationale	See Notes 1 and 2

Gap #	SF05_CNSC REGDOC 2.5.2_9.4_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	9.4 Deterministic safety analysis
Requirement Assessed	The deterministic safety analysis shall be conducted in accordance with the requirements specified in CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis.
Macro-Gap	SF05-01-16
Issue/Gap Description	Assessment against REGDOC-2.4.1 identified gaps in the deterministic safety analysis that are related to the use of legacy tools for some analysis.
Rationale	See Notes 1, 2 and 3

Gap #	SF05_CSA N290.1_4.3.1.4_16
Document ID	CSA N290.1
Article/Clause	4.3.1.4
Requirement Assessed	In order to credit (in the safety analysis) operator action to shut down (manually trip) the reactor, the design shall provide a) clear, well-defined, validated, and readily available operating procedures that identify the necessary actions; b) instrumentation in the control rooms to provide clear and unambiguous indication of the necessity for operator action; c) adequate time before operator action is required, following indication of the necessity for operator action inside the control rooms; and d) adequate time before operator action is required, following indication of the necessity for operator action outside the control rooms. Notes: 1) For new plants, adequate time is at least 30 min for operator action inside the control room and 60 min for operator action outside the control room. 2) For existing CANDU plants, adequate time is 15 min for operator action inside the control room and 30 min for operator action outside the control room.
Macro-Gap	SF05-09-16
Issue/Gap Description	For existing CANDUs, this clause defines adequate time for operator action from inside the control room as 15 min. Therefore, the credited 12 minutes for the analysis of HTS depressurization events in Appendix 3 of Part 3 of the Safety Report is not considered adequate time for operator action (Gap 1).
Rationale	See Note 2

Gap #	SF08_SF8 RT 2015_5.3_15
Document ID	SF8 RT 2015
Article/Clause	5.3
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: a) Safety related incidents, low level events and near misses
Macro-Gap	SF08-03-15
Issue/Gap Description	The Safety Report improvement project needs to capture changes in Margin Management and adverse trend in the erosion of margin in LBLOCA.
Rationale	See Note 2

Gap #	SF08_SF8 RT 2015_5.5_15
Document ID	SF8 RT 2015
Article/Clause	5.5
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: c) Maintenance, inspection and testing
Macro-Gap	SF08-08-15
Issue/Gap Description	Maintenance Backlogs were defined as needing improvement in the 2008 Bruce 3 and 4 ISR, based on a review of the backlog history. Although progress has been made on backlogs they are still identified as an area for improvement.
Rationale	In Letter from F. Saunders to M. Leblanc, ' Bruce A and Bruce B Licence Renewal-Supplemental Update', dated, November 27, 2014, NK21-CORR-00531-11711, NK29-CORR-00531-12101: Page A-21 to 23 of 47 states the following: “Bruce Power is maintaining focus on maintenance backlog reduction in support of improving equipment reliability and forced loss rates. As such, Bruce Power has recently electively adopted INPO AP-928 'Work Management Process Description' as an industry standard. This standard re-categorizes corrective maintenance and elective maintenance to critical component maintenance and deficient component maintenance. As demonstrated in figures below, Bruce Power has made extensive gains in reducing maintenance backlogs at both stations through the course of 2014. Both stations are now in line with, or exceed target backlog goals…..” Maintenance backlog is a well known improvement area that receives continuous focus and there are on-going initiatives to reduce them together with their performance respective indicators.

Gap #	SF08_SF8 RT 2015_5.6_15
Document ID	SF8 RT 2015
Article/Clause	5.6
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: d) Replacements of Structures, Systems and Components (SSCs) important to safety owing to failure or obsolescence
Macro-Gap	SF08-05-15
Issue/Gap Description	Future improvements such as updating the extent of the changes to trip coverage windows for the key aged impacted accident scenarios of loss of flow, neutron overpower protection, small break loss of coolant accidents in compliance with R-10, while considering the use of the modified 37-element bundle to reduce the trip coverage window [248] will be captured later in the Safety Report Improvement project [253].
Rationale	See Note 2. In addition, capturing improvements from design and operational changes such as updating the extent of the changes to trip coverage windows for the key aged impacted accident scenarios of loss of flow, neutron overpower protection, small break loss of coolant accidents is addressed through BP-PROC-00363 Nuclear Safety Assessment and its supporting procedures driven by DPT-NSAS-00002 Safety Report Analysis Update Overview.

Gap #	SF08_SF8 RT 2015_5.7_15
Document ID	SF8 RT 2015
Article/Clause	5.7
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: e) Modifications, either temporary or permanent, to SSCs important to safety
Macro-Gap	SF08-06-15
Issue/Gap Description	The documentation coverage for postulated initiating events not explicitly addressed in the Safety Report or PSAs needs to be improved. Neither the Safety Report deterministic safety analysis nor the PSAs explicitly include Crane Hazard analysis. Complete Hazard Analysis of Record and integrate it with the Deterministic Analysis and PSAs.
Rationale	See Note 2

Gap #	SF08_SF8 RT_5.6_16
Document ID	SF8 RT
Article/Clause	5.6
Requirement Assessed	Future improvements such as updating the extent of the changes to trip coverage windows for the key aged impacted accident scenarios of loss of flow, neutron overpower protection, small break loss of coolant accidents in compliance with R-10, while considering the use of the modified 37-element bundle to reduce the trip coverage window [169] will be captured later in the Safety Report Improvement project [174]. For completeness, this is identified as gap SF8-5 in Table 10.
Macro-Gap	SF08-04-16
Issue/Gap Description	Future improvements such as updating the extent of the changes to trip coverage windows for the key aged impacted accident scenarios of loss of flow, neutron overpower protection, small break loss of coolant accidents in compliance with R-10, while considering the use of the modified 37-element bundle to reduce the trip coverage window [248] will be captured later in the Safety Report Improvement project [253].
Rationale	See Note 2. In addition, capturing improvements from design and operational changes such as updating the extent of the changes to trip coverage windows for the key aged impacted accident scenarios of loss of flow, neutron overpower protection, small break loss of coolant accidents is addressed through BP-PROC-00363 Nuclear Safety Assessment and its supporting procedures driven by DPT-NSAS-00002 Safety Report Analysis Update Overview.

Gap #	SF11_SF11 RT_5.4_15
Document ID	SF11 RT
Article/Clause	5.4
Requirement Assessed	Review task 3 examines maintenance, testing and inspection procedures.
Macro-Gap	SF11-01-15
Issue/Gap Description	Some difficulties were identified that are related to the effectiveness of maintenance planning and scheduling. Additionally, the effectiveness of Bruce Power Preventative Maintenance Oversight Group (PMOG) to address the Preventative Maintenance (PM) backlog was assessed by CNSC staff and it was noted Bruce Power is experiencing challenges.
Rationale	In Letter from F. Saunders to M. Leblanc, ' Bruce A and Bruce B Licence Renewal-Supplemental Update', dated, November 27, 2014, NK21-CORR-00531-11711, NK29-CORR-00531-12101: Page A-21 to 23 of 47 states the following: “Bruce Power is maintaining focus on maintenance backlog reduction in support of improving equipment reliability and forced loss rates. As such, Bruce Power has recently electively adopted INPO AP-928 'Work Management Process Description' as an industry standard. This standard re-categorizes corrective maintenance and elective maintenance to critical component maintenance and deficient component maintenance. As demonstrated in figures below, Bruce Power has made extensive gains in reducing maintenance backlogs at both stations through the course of 2014. Both stations are now in line with, or exceed target backlog goals.....” Maintenance backlog is a well known improvement area that receives continuous focus and there are on-going initiatives to reduce them together with their performance respective indicators.

Gap #	SF12_CSA N290.12_4.1.2_16
Document ID	CSA N290.12
Article/Clause	4.1.2
Requirement Assessed	Plans for HF in design activities shall be revised when needed. --- Notes: 1) Plans for HF in design activities should reflect the entire project scope. 2) Examples include when more design detail is available or when project scope or strategy change.
Macro-Gap	SF12-05-16
Issue/Gap Description	Provision for the revision of HF in design activities is not identified explicitly in DPT-PDE-00013.
Rationale	This gap has been addressed in Appendix D, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_4.1.6_16
Document ID	CSA N290.12
Article/Clause	4.1.6
Requirement Assessed	Planning for HF in design shall consider constraints and drivers. --- Notes: 1) This includes a) HF-related constraints such as operating policies and principles; b) HF-related drivers such as goals for performance improvement; and c) considerations related to construction, commissioning, operation, maintenance, and decommissioning. 2) For the purposes of this Clause, "shall consider" means that the user evaluates the impact and documents any decisions (e.g., no action, operating procedures, and design features).
Macro-Gap	SF12-05-16
Issue/Gap Description	Discussion on constraints and drivers in planning for HF in design is not included in any Bruce Power Human Factors documentation.
Rationale	This gap has been addressed in Appendix J, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_4.3_16
Document ID	CSA N290.12
Article/Clause	4.3 Organization and resources
Requirement Assessed	Planning for HF in design shall define a) HF in design roles, authorities, and resources within the project organization and the HF in design reporting relationships; --- Note: Examples of authorities include preparers, reviewers, approvers, acceptors, and design authority. b) supporting resources necessary for HF in design work; and Note: Examples include intended users and simulators. c) provision for continuity. Note: This applies to lengthy projects to address potential turnover of the project team.
Macro-Gap	SF12-05-16
Issue/Gap Description	b) supporting resources necessary for HF in design work has not been identified
Rationale	This gap has been addressed in Section 7 and throughout the Appendices in Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_4.3_16
Document ID	CSA N290.12
Article/Clause	4.3 Organization and resources
Requirement Assessed	Planning for HF in design shall define a) HF in design roles, authorities, and resources within the project organization and the HF in design reporting relationships; --- Note: Examples of authorities include preparers, reviewers, approvers, acceptors, and design authority. b) supporting resources necessary for HF in design work; and Note: Examples include intended users and simulators. c) provision for continuity. Note: This applies to lengthy projects to address potential turnover of the project team.
Macro-Gap	SF12-05-16
Issue/Gap Description	c) provision for continuity for lengthy projects where turnover can be expected is not discussed
Rationale	This gap has been addressed in Appendix D, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.2.1_16
Document ID	CSA N290.12
Article/Clause	5.2.1
Requirement Assessed	HF in design should consider the scope, content, and usability of procedures in relation to SSCs being designed. --- Note: Attention should be paid to human error scenarios that might arise as a result of a new design, or as a result of differences with an older design. These should be considered as inputs for the development or revision of procedures.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no Bruce Power documentation that suggests that Bruce Power's HF program considers the scope, content, and usability of procedures in relation to SSCs being designed.
Rationale	This gap has been addressed in Appendix K, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.2.3_16
Document ID	CSA N290.12
Article/Clause	5.2.3
Requirement Assessed	HF in design should consider the following information: a) equipment manufacturer standard operating and maintenance instructions; and b) plant operating and maintenance instructions.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance on the sources of information that should be used for procedure development.
Rationale	This gap has been addressed in Appendix K, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.2.4_16
Document ID	CSA N290.12
Article/Clause	5.2.4
Requirement Assessed	HF in design should collaborate with the procedures development discipline to identify and develop procedures.
Macro-Gap	SF12-05-16
Issue/Gap Description	Currently, the documentation suggests that HF in design does not collaborate with procedure development at Bruce Power to identify and develop procedures.
Rationale	This gap has been addressed in Appendix K, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.3.1_16
Document ID	CSA N290.12
Article/Clause	5.3.1
Requirement Assessed	HF in design should review the scope, content, and timing of training in relation to new or updated tasks and systems. --- Note: This can help the HF practitioner to understand the skills and qualifications of the users.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no Bruce Power documentation that suggests that Bruce Power's HF program considers the scope, content, and timing of training in relation to new or updated tasks and systems.
Rationale	This gap has been addressed in Appendices H and K, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.3.4_16
Document ID	CSA N290.12
Article/Clause	5.3.4
Requirement Assessed	HF in design may collaborate with the training development discipline to identify and develop training.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that HF in design may collaborate with the training development discipline to identify and develop training.
Rationale	This gap has been addressed in Appendices H and K, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.4.2_16
Document ID	CSA N290.12
Article/Clause	5.4.2
Requirement Assessed	Safety analysis personnel may participate in HF in design evaluation activities.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that safety analysis personnel may participate in HF in design evaluations.
Rationale	This gap has been addressed in Appendix I, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_5.4.4_16
Document ID	CSA N290.12
Article/Clause	5.4.4
Requirement Assessed	HF in design should collaborate with the safety analysis discipline to identify and analyze human actions.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that safety analysis and HF in design should collaborate to identify and analyze human actions.
Rationale	This gap has been addressed in Appendix I, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.1.6_16
Document ID	CSA N290.12
Article/Clause	6.1.6
Requirement Assessed	HF in design should consider the impact of combining existing systems and new systems on a) human performance; and b) processes and procedures.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation to suggest that HF in design considers the impact of combining existing systems and new systems on human performance and processes and procedures.
Rationale	This gap has been addressed in Appendix N, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.2.2_16
Document ID	CSA N290.12
Article/Clause	6.2.2
Requirement Assessed	The following HF in design activities should be started during scoping and conceptual design: a) OER; b) functional analysis; c) development or selection of HF in design source documents; d) a statement of the operational purpose of the system and the operational requirements under all anticipated conditions; --- Note: This may include a description of the a) working environment; b) plant command and control philosophy; c) staffing concept with an indication of the required personnel capabilities and responsibilities; and d) human-system performance requirements. --- e) identification of scenarios to be analyzed; --- Note: Inputs to be considered include a) plant, system, and equipment states; b) degraded equipment conditions; c) human actions important to safety; d) operating experience; e) operating procedures; and f) requirements for personal protective equipment. --- f) identification of SSC requirements to support necessary human actions; and --- Notes: 1) Examples of SSC requirements include a) parameters necessary to supervise SSCs; b) parameters necessary to confirm automatic safety actions; and c) controls necessary to manually carry out operator-initiated actions. 2) Necessary human actions include specific activities within operations, maintenance, testing, repair, and inspection that are judged as required for successful outcomes. --- g) assessment of design concepts and options. --- Notes: 1) The assessment should consider human performance

	requirements, capabilities and limitations, task requirements, HSI performance requirements, and other design considerations. 2) Consideration should be given to user-configured displays, annunciations, and set-points to ensure safe and appropriate use.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is lack of detailed guidance on when HF activities should be conducted during the design process.
Rationale	This gap has been addressed in Appendix A, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.3.1_16
Document ID	CSA N290.12
Article/Clause	6.3.1
Requirement Assessed	High-level HF-related requirements shall be documented during early/preliminary design.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation to suggest when high level HF related requirements shall be documented.
Rationale	This gap has been addressed in Appendix A, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.3.2_16
Document ID	CSA N290.12
Article/Clause	6.3.2
Requirement Assessed	The following HF in design activities should be started during early/preliminary design: a) task analysis; b) modelling, mock-ups, or prototyping of user interfaces; c) evaluations; --- Notes: 1) Examples include a) usability testing; b) checking against HF in design source documents; and c) inspection-based methods, such as heuristic evaluations. 2) HF-related evaluation of vendor products should be included in the technical specification. --- d) input to specifications and bid evaluations; --- Note: Considerations include a) time necessary for procurement; b) time necessary for the evaluation of vendor submissions; c) technical exceptions; and d) equivalency evaluations for obsolete SSCs. --- e) participation in the assessment of human actions and error consequences; and --- Note: Examples of assessments include a) human reliability analysis (as part of probabilistic safety assessment); b) safety analyses; c) hazard and operability studies; d) assessments of constructability, operability, maintainability, and safety; and e) failure modes and effects analyses. --- f) assessment of the feasibility of human actions in the deterministic safety analyses.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is lack of detailed guidance on when HF activities should be conducted during the design process.
Rationale	This gap has been addressed in Appendix A, Rev09 of DPT-PDE-00013

Gap #	SF12_CSA N290.12_6.3.3_16
Document ID	CSA N290.12
Article/Clause	6.3.3
Requirement Assessed	The following activities should be completed during early or preliminary design: a) OER; b) development or selection of HF in design source documents; c) functional analyses; and d) a statement of the operational purpose of the system and the operational requirements under all anticipated conditions.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is lack of detailed guidance on when HF activities should be conducted during the design process.
Rationale	This gap has been addressed in Appendix A, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.4.1_16
Document ID	CSA N290.12
Article/Clause	6.4.1
Requirement Assessed	The following HF in design activities shall be completed during advanced or detailed design: a) detailed HSI design; b) confirmation of the feasibility of human actions important to safety in the probabilistic and deterministic safety analyses; and c) where applicable, design integration of COTS products.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is lack of detailed guidance on when HF activities should be completed during the design process.
Rationale	This gap has been addressed in Appendix A, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.4.2_16
Document ID	CSA N290.12
Article/Clause	6.4.2
Requirement Assessed	The following HF in design activities should be completed during advanced or detailed design: a) analyses to confirm the ability of the human to perform necessary actions; b) usability testing; c) verification; --- Note: Verification should be carried out before the design is released for construction. --- d) validation; and --- Note: While validation is important during design, validation activities could be split between detailed design and implementation. --- e) output of HF in design analyses for the development of training manuals, operating procedures, and commissioning procedures.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is lack of detailed guidance on when HF activities should be completed during the design process.
Rationale	This gap has been addressed in Appendix A, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.5.3_16
Document ID	CSA N290.12
Article/Clause	6.5.3
Requirement Assessed	Evaluation of the as-built design shall be performed. --- Notes: 1) Examples include equipment location, accessibility, panel layouts, instrument configuration, identification, and indication. 2) The HF in design plan should specify the need for participation in a walk-down. ---
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that evaluation of the as-built design shall be performed.
Rationale	This gap has been addressed in Appendix L, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_6.5.4_16
Document ID	CSA N290.12
Article/Clause	6.5.4
Requirement Assessed	HF deficiencies identified during commissioning shall be addressed prior to declaring the system available for service. --- Note: Deficiencies may be addressed by a) changing the design; b) correction of the installation; or c) provision of a disposition.
Macro-Gap	SF12-05-16
Issue/Gap Description	Although Bruce Power documentation suggests that HF issues be addressed before DCN and DCP close out, the requirements for engineering change close – out as defined in BP-PROC-00539 are not until after the declaration of AFS.
Rationale	This gap has been addressed in Section 4.4.4 & 4.4.5, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_7.1_16
Document ID	CSA N290.12
Article/Clause	7.1
Requirement Assessed	The rationale for the selected analysis techniques shall be documented. --- Notes: 1) Examples of analysis techniques (see Annex B) are a) OER; b) functional analysis; c) task analysis, as follows: i) human error analysis; ii) workload analysis; iii) physical demands analysis; and iv) communications analysis; and d) link analysis. 2) The output of one analysis may be used as input to another analysis.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that the rationale for the selected analysis techniques shall be documented.
Rationale	This gap has been addressed in individual appendixes in DPT-PDE-00013 R09 outlining the review elements all include this requirement.

Gap #	SF12_CSA N290.12_8.11_16
Document ID	CSA N290.12
Article/Clause	8.11
Requirement Assessed	For evaluations of COTS products, or where a largely pre-developed design is being considered, the HF in design should define the HF-related requirements and criteria against which the design is to be evaluated. --- Note: Examples of bases for requirements and criteria include a) analysis of the adequacy of the HF in design work carried out by the vendor; b) establishing user requirements for the product, including transfer of training issues; c) analysis of tasks and intended context of use for the COTS product; d) consideration of interfaces of the product with other plant systems; e) identification of usability, safety impact, and human performance issues in the anticipated contexts of use; and f) impacts on maintenance, training, and procedures.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation to suggest that HF in design should define HF-related requirements and criteria against which the design is to be evaluated.
Rationale	This gap has been addressed in Appendices L, M and N, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_8.12_16
Document ID	CSA N290.12
Article/Clause	8.12
Requirement Assessed	Where COTS products are proposed and there is foreseeable impact on HF, evaluation of the potential COTS options should be carried out.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation to suggest that HF in design should evaluate potential COTS options.
Rationale	This gap has been addressed in Appendix T, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_8.5_16
Document ID	CSA N290.12
Article/Clause	8.5
Requirement Assessed	Evaluations should be conducted throughout the project. --- Notes: 1) Evaluations can provide data to guide the detailed design. 2) Where appropriate, certain HF-related evaluations can be performed during installation and commissioning.
Macro-Gap	SF12-05-16
Issue/Gap Description	There is no documentation or guidance to suggest that evaluations should be conducted throughout the project.
Rationale	This gap has been addressed in Appendix M, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_8.6_16
Document ID	CSA N290.12
Article/Clause	8.6
Requirement Assessed	The following evaluation approaches (see Annexes C and D) may be used: a) usability trials; b) inspection-based evaluations; c) verification; and d) validation.
Macro-Gap	SF12-05-16
Issue/Gap Description	While verification and validation are described well in DPT-PDE-00013, there is no guidance to suggest that usability trials and inspection-based evaluations should be conducted.
Rationale	This gap has been addressed in Validation section, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_8.8_16
Document ID	CSA N290.12
Article/Clause	8.8
Requirement Assessed	Where available, the evaluation test subjects should be independent from the design team for the final and late-stage evaluations.
Macro-Gap	SF12-05-16
Issue/Gap Description	The independence of subject matter experts that may be recruited to be involved in validation exercises is not explicitly discussed in any Bruce Power HF documentation.
Rationale	This gap has been addressed in Appendix M, Rev09 of DPT-PDE-00013.

Gap #	SF12_CSA N290.12_8.9_16
Document ID	CSA N290.12
Article/Clause	8.9
Requirement Assessed	An HF practitioner independent from the design team should review the plans for the final or late-stage evaluations.
Macro-Gap	SF12-05-16
Issue/Gap Description	The review of the plans for final or late-stage evaluations by an HF practitioner independent from the design team is not explicitly discussed in any Bruce Power HF documentation.
Rationale	DPT-PDE-00013 (R09) is updated to account for independent reviews on an as required basis in Appendix M.

Gap #	SF12_NUREG-0700_Part_I_15
Document ID	NUREG-0700
Article/Clause	Part_I
Requirement Assessed	Part I of NUREG-0700 provides guidelines for the basic HSI elements: information display, user interface interaction and management and controls
Macro-Gap	SF12-02-15
Issue/Gap Description	Field components, particularly those that would be referenced in emergency procedures (i.e. AIMS), were not reviewed against NUREG-0700 or any modern standards or guidelines.
Rationale	DPT-PDE-00013 (R09) is updated to account for reviews using internal and external guidelines including NUREG-0700 and provides a hierarchy of priority under normal conditions. Appendix J shows a list of inputs into Human System interfaces and design guides are part of this. Appendix T (2) discusses the general hierarchy of these inputs. Appendix Q specifically notes NUREG-0700 as "additional" design guidance.

Gap #	SF12_NUREG-0700_Part_I_16
Document ID	NUREG-0700
Article/Clause	Part_I
Requirement Assessed	Part I of NUREG-0700 provides guidelines for the basic HSI elements: information display, user interface interaction and management and controls
Macro-Gap	SF12-04-16
Issue/Gap Description	Field components, particularly those that would be referenced in emergency procedures (i.e. AIMS), were not reviewed against NUREG-0700 or any modern standards or guidelines.
Rationale	DPT-PDE-00013 (R09) is updated to account for reviews using internal and external guidelines including NUREG-0700 and provides a hierarchy of priority under normal conditions. Appendix J shows a list of inputs into Human System interfaces and design guides are part of this. Appendix T (2) discusses the general hierarchy of these inputs. Appendix Q specifically notes NUREG-0700 as "additional" design guidance.

Gap #	SF12_NUREG-0700_Part_II_4_15
Document ID	NUREG-0700
Article/Clause	Part_II_4
Requirement Assessed	Guidelines for reviewing alarm system.
Macro-Gap	SF12-02-15
Issue/Gap Description	Lack of design guidance for Bruce A annunciation systems
Rationale	DPT-PDE-00013 (R09) is updated to account for design guidance for annunciation systems.

Gap #	SF13_CNSC REGDOC 2.10.1_2.1_15
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.1 Planning basis
Requirement Assessed	All licensees shall: 1. establish a planning basis for their EP program 2. ensure the planning basis considers the hazards that have, or could have, an adverse impact on the environment and the health and safety of onsite personnel or the public, and also consider: a. all accidents and internal or external events that have been analyzed as having an unacceptable impact on their facilities b. the inclusion of multi-unit accidents scenarios for multi-unit power reactor facilities c. extended loss of power 3. use the results from the planning basis to determine the scope and depth of EP program requirements Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW. These licensees shall: 4. provide regional and provincial offsite authorities with necessary information to allow for effective emergency planning policies and procedures to be established and modified, if needed, periodically Guidance Guidance for all licensees A nuclear emergency may be caused by, or involve, different types of hazards, including natural incidents (e.g., flooding, tornadoes, tsunami, ice or snowstorms, forest fires) and equipment malfunctions (identified within the design basis and beyond design basis). All hazards that cannot be practically eliminated with possible initiating and propagating pathways should be identified within the planning basis. Response to criminal and malicious activity may be dealt with under a separate program. The planning basis should be based on a full range of postulated scenarios that may challenge the facility's emergency response capabilities. This should include scenarios that involve a nuclear or radiological emergency combined with a conventional emergency, such as an earthquake or forest fire. A detailed analysis may be used to determine scenarios that can be practically eliminated. Plans should be developed for those scenarios that cannot be practically eliminated. Inputs to be considered in the analysis should include: the licensee's safety analysis, probabilistic safety analysis, and operating experience. Additional guidance for licensees of reactor facilities with a thermal

	capacity greater than 10 MW The information to be provided to regional and provincial offsite authorities should give all necessary details to make informed decisions on the size of emergency planning zones and the level of preparedness required. The necessary information should include: • possible accidents that cannot be practically eliminated • an estimate of the probability of such accidents occurring • an estimate of the associated radiological consequences, including isotopic release quantities, possible release start time and duration and the geographical area potentially affected Federal authorities would be provided emergency planning information through the CNSC.
Macro-Gap	SF13-02-15
Issue/Gap Description	Completion and/or resolution of Fukushima Action Items, which includes: • Completion of SAMG updates to provide guidance for multi-unit severe accidents;
Rationale	This gap is already covered under letter from F. Saunders to K. Lafreniere, Integrated Implementation Plan for Bruce A, Bruce B and Center of Site in the Next Licence Period, NK21-CORR-00531-11567, NK29-CORR-00531-11950, NK37-CORR-00531-02288, dated October 31, 2014. “Implement Enhancements to SAMG' is related to CNSC AI 1307-3703 CNSC Review of Bruce Power Multi-Unit Severe Accident Modeling and Plan for Model Improvement (FAI 3.2.1 and 3.2.2). Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 7 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12209 dated August 7, 2015 which states the following in Attachment B, Section 2.9: ‘FAI 3.2.1 and 3.2.2 were closed by the CNSC in .... The project is completed.....The results of the work indicate that scaling and injection methods used previously to approximate multi-unit accidents in the single-unit MAAP4-CANDU models agree with the newly developed multi-unit model. The predictions from these two approaches are sufficiently well aligned such that further development of multi-unit models for Bruce A and Bruce B is not warranted..... Bruce Power requests closure of AI 1307-3703. AI 1307-3703 is Closed based on the latest Fukushima Action Plan update: Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 9 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12828/NK29-CORR-00531-13279/ NK37-CORR-00531-02560 dated June 26, 2016

Gap #	SF13_CNSC REGDOC 2.10.1_2.1_16
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.1 Planning basis
Requirement Assessed	All licensees shall: 1. establish a planning basis for their EP program 2. ensure the planning basis considers the hazards that have, or could have, an adverse impact on the environment and the health and safety of onsite personnel or the public, and also consider: a. all accidents and internal or external events that have been analyzed as having an unacceptable impact on their facilities b. the inclusion of multi-unit accidents scenarios for multi-unit power reactor facilities c. extended loss of power 3. use the results from the planning basis to determine the scope and depth of EP program requirements Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW. These licensees shall: 4. provide regional and provincial offsite authorities with necessary information to allow for effective emergency planning policies and procedures to be established and modified, if needed, periodically Guidance Guidance for all licensees A nuclear emergency may be caused by, or involve, different types of hazards, including natural incidents (e.g., flooding, tornadoes, tsunami, ice or snowstorms, forest fires) and equipment malfunctions (identified within the design basis and beyond design basis). All hazards that cannot be practically eliminated with possible initiating and propagating pathways should be identified within the planning basis. Response to criminal and malicious activity may be dealt with under a separate program. The planning basis should be based on a full range of postulated scenarios that may challenge the facility's emergency response capabilities. This should include scenarios that involve a nuclear or radiological emergency combined with a conventional emergency, such as an earthquake or forest fire. A detailed analysis may be used to determine scenarios that can be practically eliminated. Plans should be developed for those scenarios that cannot be practically eliminated. Inputs to be considered in the analysis should include: the licensee's safety analysis, probabilistic safety analysis, and operating experience. Additional guidance for licensees of reactor facilities with a thermal

	capacity greater than 10 MW The information to be provided to regional and provincial offsite authorities should give all necessary details to make informed decisions on the size of emergency planning zones and the level of preparedness required. The necessary information should include: • possible accidents that cannot be practically eliminated • an estimate of the probability of such accidents occurring • an estimate of the associated radiological consequences, including isotopic release quantities, possible release start time and duration and the geographical area potentially affected Federal authorities would be provided emergency planning information through the CNSC.
Macro-Gap	SF13-01-16
Issue/Gap Description	Severe Accident Management Guidance (SAMG) implementation that will input the planning basis to cater to a wider range of multi-unit severe accidents is in progress. Also, an upgrade to the Emergency Response Plan (ERP) code to allow multi-unit dose projection modeling capability remains in progress and should be in-service by Q1 of 2017.
Rationale	This gap is already covered under Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 9 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12828/ NK29-CORR-00531-13279/NK37-CORR-00531-02560: Bruce Power is in the process of upgrading its Emergency Response Projection (ERP) code to allow multi-unit dose projection modeling capability. This work is being undertaken jointly with Ontario Power Generation and is targeted for completion in 2017. The status of the project is tracked under AI 1307-3790.

Gap #	SF13_CNSC REGDOC 2.10.1_2.2.3_15
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.2.3 Emergency assessment requirements
Requirement Assessed	All licensees shall: In accordance with ER plans and procedures: 1. describe the methods and procedures to continually assess the emergency and predict both onsite and offsite conditions and parameters 2. continuously take appropriate measures to protect onsite personnel 3. continually characterize the magnitude of the offsite risk to the public and the environment 4. continually provide updates on a regular basis to offsite authorities and the CNSC Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW. These licensees shall: 5. have real-time fixed radiological detection and monitoring capabilities around the nuclear facility perimeter with appropriate backup power, and shall communicate results to offsite authorities and the CNSC 6. have sufficient capacity and capability for offsite radiological monitoring, including mobile offsite survey teams, and report results to the offsite response authorities and the CNSC 7. promptly and continuously assess and determine source term estimate, plume dispersion and dose modeling, and report results to the offsite authorities and the CNSC 8. promptly and continuously estimate dose to the public based on source term estimation, plume dispersion and dose modeling, and provide the dose estimates to offsite response authorities and the CNSC Guidance Guidance for all licensees Emergency assessment, including categorization, is performed to determine: • the onsite response and staff mobilization required to protect onsite personnel and equipment • the notification category necessary for the provincial or territorial authorities to determine the required offsite response to protect the public and the environment Licensees should describe the methods and procedures for continual assessment of the following pertinent conditions and parameters: • the status, integrity and stability of the affected facilities and their

	components • identification, quantities, concentrations, or release rates of radiation, contaminants or other hazardous substances • onsite and offsite impacts on or threats to health, safety and the environment • location and direction of radioactive plumes or other emissions • loss of instrumentation Additional guidance for licensees of reactor facilities with a thermal capacity greater than 10 MW Source term sampling and estimation should be determined and reported to the CNSC on an hourly basis, upon determination and compilation of the data in a format approved by the provincial authority.
Macro-Gap	SF13-01-15
Issue/Gap Description	General improvements/revisions to the Emergency Measures Program, the BPNERP, and implementing documents is required, which includes: • implementation of real-time off-site fixed radiological detection and monitoring;
Rationale	This is covered under IIP-2014 under GIO-011 “Implement Enhancements to SAMG”. Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 7 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12209 dated August 7, 2015 states the following: ‘In Reference B7, Bruce Power notified the CNSC that the installation of remote monitoring equipment, including 44 gamma detectors (16 on-site detectors with the remaining 28 within the 10 km area around the site) was complete. CNSC staff closed AI 1307-3797 in Reference B15. Additional work, which goes beyond the scope of the original requirements is being completed to install 8 air particulate monitors in Q4 of 2015. These air samplers will augment the existing Bruce Power Environmental Tritium air monitors in order to provide more detailed data in terms of airborne and ground deposition. Installation of the additional air particulate monitors remains on track; however there is a risk that installation plans may be disrupted by winter weather conditions.’ AI 1307-3797 is Closed based on the latest Fukushima Action Plan update: Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 9 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12828/NK29-CORR-00531-13279/ NK37-CORR-00531-02560 dated June 26, 2016

Gap #	SF13_CNSC REGDOC 2.10.1_2.3.4_15
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.3.4 Public preparedness requirements
Requirement Assessed	All licensees shall: Incorporate information on public emergency preparedness into their public information program (established as per RD/GD-99.3, Public Information and Disclosure) to ensure information on emergency preparedness and response is communicated to surrounding communities and stakeholders. Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW and with designated offsite emergency planning zones. These licensees shall provide the necessary resources and support to provincial and municipal authorities in implementing the provincial and municipal plans to do the following, or shall do the following: 1. ensure that a sufficient quantity of iodine thyroid-blocking (ITB) agents is pre-distributed, to all residences, businesses and institutions within the designated plume exposure planning zone, together with instructions on their proper administration 2. ensure that a sufficient quantity of ITB agent is pre-stocked and ready for prompt distribution within the designated ingestion control planning zone; this inventory of ITB agents shall be located so that it can be efficiently obtained by, or distributed to, members of the public when required 3. ensure that ITB agents can be obtained by residents of the designated ingestion control planning zone at any time 4. ensure that particular consideration is given to sensitive populations such as children and pregnant women within the designated ingestion control planning zone 5. ensure that the pre-distributed and pre-stocked ITB agents are maintained within expiry date 6. ensure that the pre-distribution plans are supported by a robust, ongoing, and cyclical public education program 7. ensure that all residences, businesses and institutions within the designated plume exposure planning zone are provided with public emergency preparedness information detailing how they should prepare for a nuclear emergency and what they should do or expect during a nuclear emergency; this information will reinforce the public education program designed to support the pre-distribution of ITB agents 8. ensure that this public emergency preparedness information is readily available to the general public, including online Guidance

	Guidance for all licensees Licensees may, where possible, leverage existing communication channels (such as those used by local municipalities or those identified in their public information program as per RD/GD-99.3, Public Information and Disclosure). Licensees should periodically assess the adequacy of public emergency preparedness information. Additional guidance for licensees of reactor facilities with a thermal capacity greater than 10 MW For reactor facilities with a thermal capacity greater than 10 MW and with designated offsite emergency planning zones: The term ITB agent is used generically and includes potassium iodide (KI) tablets. The pre-distribution of ITB agents should be undertaken by representatives of the health and/or emergency management authorities of the province or region/municipality, with support from the licensee. The pre-distribution of ITB agents should be done in a carefully planned and coordinated manner, to ensure that the public receives the appropriate information and education related to the benefits, risks and usage instructions of ITB agents. Pre-stocked ITB agents for the designated ingestion control planning zone should be located to facilitate prompt and efficient distribution during an emergency. Recognizable locations with credible persons within the community (such as fire stations, police stations and pharmacies) should be considered in the selection of pre-stocking locations. Following the completion of pre-distribution activities, periodic reviews with the local populations to assess the adequacy of pre-distribution programs should be performed. The term “designated plume exposure planning zone” is sometimes referred to as “primary zone”, “urgent protective action zone” or “emergency planning zone”. The size of the plume exposure planning zone is determined by the appropriate offsite authorities based on information in the planning basis and is typically sized in the range of 8 to 16 km. The term “designated ingestion control planning zone” is sometimes referred to as “secondary zone”, “extended planning distance” or “ingestion planning zone”. Appropriate offsite authorities determine the size of the ingestion control planning zone (typically in the range of 50 to 80 km) based on information in the planning basis.

	To ensure the public have easy access to the required emergency preparedness information, licensees should collaborate with municipalities to provide residents with useful information on how they should prepare, what they should expect and how they should respond to an emergency at the nuclear facility. An emergency preparedness information product should be distributed in hard copy annually to every residence, business and institution within the plume exposure planning zone, and posted on a variety of websites, including those of the licensees, municipalities and provincial EMOs. This should include information on: • how they will be alerted • how they will be notified or informed on what to do • sheltering-in-place instructions • evacuation orders • how/when to take ITB agents, and where to get them if not pre-distributed • contact details for where to obtain additional information, such as websites and social media sites Licensees may, where possible, leverage existing communication channels (such as those used by local municipalities or those identified in the public information program). In discussion with local authorities, licensees should consider providing public preparedness information with ITB packages when distributing to local populations.
Macro-Gap	SF13-01-15
Issue/Gap Description	General improvements/revisions to the Emergency Measures Program, the BPNERP, and implementing documents is required, which includes: .... • Pre-distribution of Iodine Thyroid Blocking agents requires to be implemented (committed to CNSC by year end 2015).
Rationale	Page 181 of the Minutes of the Canadian Nuclear Safety Commission (CNSC) Meeting held September 30 and October 1, 2015 states the following: 129. The Bruce Power representative stated that Bruce Power had completed the KI pre-distribution and pre-stocking as specified in its LCH and REGDOC-2.10.1. The Bruce Power representative added that Bruce Power was working on additional nuclear emergency preparedness good-practice activities with local communities. This initiative is completed based on the above Minutes and hence gap is considered Closed.

Gap #	SF13_CNSC REGDOC 2.3.2_3.3_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	3.3 Equipment and instrumentation requirements
Requirement Assessed	Licensees shall: 1. provide adequate capabilities to preserve the physical barriers for release of radioactivity and to ensure that means are available to: a. control challenges posed by DBAs within appropriate limits b. mitigate consequences of BDBAs c. reduce radiation risks from possible releases of radioactive materials by carrying out accident management actions 2. address the information needs for accident management, by providing adequate instrumentation that is capable of: a. diagnosing that an accident, including a severe accident, is occurring or has occurred b. obtaining information, as necessary, on key parameters (which may include neutron flux, temperatures, pressures, flows, combustible gas concentrations, and radiation levels) to assess accident conditions and progression c. addressing continuously the state of essential safety functions, including reactor core monitoring, reactivity control, fuel cooling, hydrogen control, and containment d. confirming the effectiveness of the accident management actions 3. demonstrate with reasonable assurance that the equipment and instruments used in severe accident management will survive and perform their intended functions in the ensuing harsh conditions
Macro-Gap	SF13-02-15
Issue/Gap Description	2., 3., For DBAs, the EQ program EQAs confirm instrumentation survivability to assess the need for and effectiveness of accident management actions. These also be credited for many BDBAs, including severe accidents. This is to be confirmed with site-specific assessments per NK21-CORR-00531-11379. However, a particular area that requires attention is the need for combustible gas concentration measurement during severe accidents. This is considered a gap. Completion and/or resolution of Fukushima Action Items, which includes: • direct measurement combustible gas concentration or acceptable resolution of issue. (Note: resolution of FAIs are progressing according to schedule acceptable to CNSC).
Rationale	This gap is already covered under letter from F. Saunders to K. Lafreniere, Integrated Implementation Plan for Bruce A, Bruce B and Center of Site in the Next Licence Period, NK21-CORR-00531-11567, NK29-CORR-00531-11950, NK37-CORR-00531-02288, dated October 31, 2014. See GIO-011 titled “Implement enhancements to SAMG” SIP-11: Fukushima Response - Severe Accident Management Enhancements

	The purpose of this initiative is to enhance the existing understanding of severe accident phenomena and SAMG capabilities. This project has a generic component, undertaken under COG Joint Project 4426 followed by station-specific implementation at each station. The scope of the work involves the following: • Enhancement of SAMG to include multi unit events and IFB events. • Assessment of instrument and equipment survivability under severe accident and identification of equipment upgrades required. • Assessment of plant habitability under severe accident conditions and identification of modifications required. • Improvement to understanding of severe accident phenomena including containment integrity, hydrogen production, aerosol behaviour, and in vessel retention. References: NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 7 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12209 dated August 7, 2015, in Attachment B, Section 2.8 Implementation of Severe Accident Management (FAI 3.1.2 to 3.1.3), states the following: The industry undertook a COG joint project to update the Generic Severe Accident Management Guidelines (SAMG) to reflect the most recent lessons learned. The scope of work included assessments of: Challenges to containment such as hydrogen The results of these assessments were documented in over twenty topical reports. The Bruce Power site specific SAMG documentation was updated to be consistent with the new COG generic SAMG documents, the COG SAMG Technical Basis Documents and the knowledge obtained from COG topical reports. In addition to the aforementioned, the following changes were also incorporated: Development of revised SAMG documentation for Computational Aid #4 (CA4)-Hydrogen Flammability in Containment CA4 constitutes an acceptable resolution of the issue. FAI 3.1.2 and 3.1.3 is Closed based on the latest Fukushima Action Plan update: Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 9 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12828/NK29-CORR-00531-13279/NK37-CORR-00531-02560

	dated June 26, 2016 This gap is considered Closed based on the above correspondence.

Gap #	SF13_CNSC REGDOC 2.3.2_3.4_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	3.4 Requirements for procedures and guidelines
Requirement Assessed	Licensees shall: 1. develop, verify and validate accident management procedures and guidelines, including EOPs and SAMGs 2. account for factors specific to the reactor design in the development of SAMGs for severe accidents 3. consider that information available to the operating staff or emergency groups may be incomplete and characterized by significant uncertainties 4. include the following in SAMGs: a. the parameters and their thresholds that define the transition from EOPs to SAMGs b. key parameters to diagnose the state of various reactor and reactor systems throughout the progression of the accident c. actions to be taken to counter the damage mechanisms that would potentially challenge the integrity of the containment, irrespective of predicted frequencies of occurrence for those damage mechanisms d. indicators that can be used to judge the success of the implemented actions e. the communication protocol to be followed during implementation of accident management f. guidance on dealing with multi-unit damage, uncovered fuel in spent fuel pools, releases of radioactive materials and hydrogen into buildings adjacent to the containment 5. ensure the EOPs and SAMGs consider sufficiently long time periods to initiate and complete required actions, taking into account the human and organizational performance and the possibility of prolonged time required to restore power due to multi-unit damage or large-scale external disturbances 6. include necessary steps into guidelines for events where supplementary equipment (also called emergency mitigating equipment (EME)) and where external supports are required to mitigate the accident consequences 7. provide for transition from the accident management activities to accident recovery
Macro-Gap	SF13-02-15
Issue/Gap Description	3. Instrumentation and equipment survivability assessment to be completed will provide insights into information available to staff (NK21-CORR-00531-11379). These are considered a gap. .... (Note: resolution of FAIs are progressing according to schedule acceptable to CNSC).
Rationale	This gap is already covered under letter from F. Saunders to K. Lafreniere, Integrated Implementation Plan for Bruce A, Bruce B and Center of Site in the Next Licence Period, NK21-CORR-00531-11567, NK29-CORR-00531-

	11950, NK37-CORR-00531-02288, dated October 31, 2014. See GIO-011 titled “Implement enhancements to SAMG”. SIP-11: Fukushima Response - Severe Accident Management Enhancements. The purpose of this initiative is to enhance the existing understanding of severe accident phenomena and SAMG capabilities. This project has a generic component, undertaken under COG Joint Project 4426 followed by station-specific implementation at each station. The scope of the work involves the following: • Enhancement of SAMG to include multi unit events and IFB events. • Assessment of instrument and equipment survivability under severe accident and identification of equipment upgrades required. • Assessment of plant habitability under severe accident conditions and identification of modifications required. Improvement to understanding of severe accident phenomena including containment integrity, hydrogen production, aerosol behaviour, and in vessel retention. Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 7 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12209 dated August 7, 2015 which states the following in Attachment B, Section 2.4 Instrumentation and Equipment Survivability (FAI 1.8.1): ‘FAI 1.8.1 was closed by the CNSC in Reference B12, based on the completion of the COG generic methodology for performing survivability assessments in CANDU nuclear power plants. The Bruce specific instrument and equipment (I&E) survivability assessment was completed, and the summary report was provided as Enclosure 2 of Reference B7.

Gap #	SF13_CNSC REGDOC 2.3.2_3.5_15
Document ID	CNSC REGDOC 2.3.2
Article/Clause	3.5 Requirements for human and organizational performance
Requirement Assessed	Licensees shall: 1. establish the organizational infrastructure necessary for implementing IAMPs, which covers aspects such as authority, organization, co-ordination of the response, plans and procedures, training, drills and exercises, human factors, and quality assurance programs. 2. ensure that personnel involved in managing an accident have the necessary information, procedures, and human and materiel resources to carry out effective accident management and mitigation actions 3. clearly define the roles, responsibilities and authorities for the personnel involved in accident management and ensure coordination among different organizations 4. ensure that the IAMP contains provisions for the setup of emergency response facilities 5. establish and implement initial and continuing training programs for all personnel who are required to respond to accidents in accordance with the principles of a systematic approach to training 6. make sufficient provisions to ensure habitability of facilities required to support human performance during the implementation of the IAMP or provide alternate habitable facilities
Macro-Gap	SF13-02-15
Issue/Gap Description	Completion and/or resolution of Fukushima Action Items, which includes: .... • Completion of required studies (e.g., instrumentation and equipment survivability, in vessel retention, shield tank overpressure protection, plant habitability); .... (Note: resolution of FAIs are progressing according to schedule acceptable to CNSC).
Rationale	This gap is already covered under letter from F. Saunders to K. Lafreniere, Integrated Implementation Plan for Bruce A, Bruce B and Center of Site in the Next Licence Period, NK21-CORR-00531-11567, NK29-CORR-00531-11950, NK37-CORR-00531-02288, dated October 31, 2014. See GIO-011 titled “Implement enhancements to SAMG”. SIP-11: Fukushima Response - Severe Accident Management Enhancements The purpose of this initiative is to enhance the existing understanding of severe accident phenomena and SAMG capabilities. This project has a generic component, undertaken under COG Joint Project 4426 followed by station-specific implementation at each station. The scope of the work involves the following: • Enhancement of SAMG to include multi unit events and IFB events.

	• Assessment of instrument and equipment survivability under severe accident and identification of equipment upgrades required. • Assessment of plant habitability under severe accident conditions and identification of modifications required. Improvement to understanding of severe accident phenomena including containment integrity, hydrogen production, aerosol behaviour, and in vessel retention. References: NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 Letter from F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 7 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12209 dated August 7, 2015 which states the following in Attachments A and B with respect to the following: • Instrumentation and equipment survivability- Section 2.4 of Attachment B states that FAI 1.8.1 was closed by the CNSC (CNSC Review of Bruce Power Update#4 on Fukushima - New Action Item 2014-07-3688", May 5, 2014, EDOC# 4422724, NK21-CORR-00531-11298 / NK29-CORR-00531-11708 /NK37-CORR-00531-02229), based on the completion of the COG generic methodology for performing survivability assessments in CANDU nuclear power plants. • In vessel retention- Section 2.8 of Attachment B states that assessment of IVR has been completed. • Shield tank overpressure protection-In progress per AI 2014-07-3688. TCD-Q3 2016 • Plant habitability- Section 2.5 of Attachment B states that the station specific habitability assessment indicated that Bruce Power's installed and planned upgrades and additional lines of defence are sufficient to terminate event progression at or before early In Vessel Retention stage. Based on the assessments completed CNSC staff closed FAI 1.9.1 (Letter, K. Lafreniere to F. Saunders "Bruce Power Progress Report No. 6 on CNSC Action Plan-Fukushima Action Items", February 13,2015, NK21-CORR-00531-11940 / NK29-CORR-00531-12323/ NK37-CORR-00531-02372) Shield tank overpressure protection-In progress per AI 2014-07-3688. Latest Fukushima update (F. Saunders to K. Lafreniere, Bruce Power Progress Report No. 9 on CNSC Action Plan - Fukushima Action Items, NK21-CORR-00531-12828/NK29-CORR-00531-13279/NK37-CORR-00531-02560 dated June 26, 2016): Bruce A&B Target Completion form installation is Q2-2019.

Gap #	SF13_CSA N1600-14_4.6.1_15
Document ID	CSA N1600-14
Article/Clause	4.6.1 Nuclear emergency recovery plan development
Requirement Assessed	Requirements for the development and content of nuclear emergency recovery plans.
Macro-Gap	SF13-04-15
Issue/Gap Description	Addressing the additional requirements in CSA N1600-14. There are a number of detailed additional requirements in CSA N1600-14 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: .... • detailed requirements for nuclear emergency recovery plans. Given that CSA N1600-14 is likely to be substantially revised in the short term, a phased approach should be taken to its detailed review for elements that need to be addressed by Bruce Power.
Rationale	Transition Plan in place. LCH required Bruce power to submit to the CNSC by June 30, 2015, a transition plan for compliance with REGDOC-2.10.1. This was transmitted in NK29-CORR-00531-12566, dated June 29, 2015 which stated Bruce Power would be in full compliance by August 31, 2018, and which included the development of a Bruce Power Recovery Plan. As discussed, the 2106 corporate exercise is to include a test of the recovery plan concept.

Gap #	SF13_CSA N1600-14_4.6.1_16
Document ID	CSA N1600-14
Article/Clause	4.6.1
Requirement Assessed	Requirements for the development and content of nuclear emergency recovery plans.
Macro-Gap	SF13-02-16
Issue/Gap Description	CSA N1600 has more detailed requirements for the development and content of nuclear emergency recovery plans in comparison to CNSC REGDOC-2.10.1.
Rationale	Transition Plan in place. LCH required Bruce power to submit to the CNSC by June 30, 2015, a transition plan for compliance with REGDOC-2.10.1. This was transmitted in NK29-CORR-00531-12566, dated June 29, 2015 which stated Bruce Power would be in full compliance by August 31, 2018, and which included the development of a Bruce Power Recovery Plan. As discussed, the 2106 corporate exercise is to include a test of the recovery plan concept.

Appendix G – CATEGORY 4: Safety Improvement Considered Necessary

Appendix G consists of those micro-gaps identified in the Safety Factor Reports for which safety improvements are considered necessary.

- Table 54 provides a consolidation of all micro-gaps within this category. It is ordered such that gaps that are similar or identical appear consecutively. This can be regarded as a “smart table of contents” for the micro-gaps discussed in the next bullet, and provides a direct linkage back to the origin of the micro-gaps in the Safety Factor Reports.
- Table 55 provides the details for each of the micro-gaps within this category. This is based on an export from the PSR database, and is ordered first by Safety Factor, then by regulatory document/code/standard, then by clause.

The micro-gap number, which is provided in both tables, facilitates their use.

Table 54: Consolidation of Micro-gaps for Which Safety Improvements are Considered Necessary

Category 4- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF01_ANSI/NIRMA CM 1.0-2007_3.2_16	SF01-24-16	ANSI/NIRMA CM 1.0-2007	3.2	1
SF01_CNSC REGDOC 2.5.2_8.13_16	SF01-14-16	CNSC REGDOC 2.5.2	8.13	46
SF01_CNSC REGDOC 2.5.2_8.13_15	SF01-14-15	CNSC REGDOC 2.5.2	8.13	46
SF01_CNSC REGDOC 2.5.2_8.13.1_15	SF01-14-15	CNSC REGDOC 2.5.2	8.13.1	47
SF01_CNSC REGDOC 2.5.2_8.13.1_16	SF01-14-16	CNSC REGDOC 2.5.2	8.13.1	47
SF01_CNSC REGDOC 2.5.2_8.13.1_15	SF01-14-15	CNSC REGDOC 2.5.2	8.13.1	48
SF01_CNSC REGDOC 2.5.2_8.13.1_16	SF01-14-16	CNSC REGDOC 2.5.2	8.13.1	48
SF01_CSA N289.1_3.1_16	SF01-16-16	CSA N289.1	3.1	55
SF03_CSA N289.1_5_16	SF03-05-16	CSA N289.1	5	55
SF01_CSA N290.0-11_4.5-4.8_15	SF01-05-15	CSA N290.0-11	4.5-4.8	58
SF01_CSA N290.1_4.2.1.1_15	SF01-05-15	CSA N290.1	4.2.1.1	58
SF08_SF8 RT 2015_5.8_15	SF08-09-15	SF8 RT 2015	5.8	58
SF01_CSA N290.1_4.7.2_16	SF01-12-16	CSA N290.1	4.7.2	62
SF01_CSA N290.1_4.7.2_15	SF01-12-15	CSA N290.1	4.7.2	62
SF01_CSA N290.11-13_5.2.2.4_15	SF01-07-15	CSA N290.11-13	5.2.2.4	63
SF01_CSA N290.11-13_5.2.2.4_16	SF01-07-16	CSA N290.11-13	5.2.2.4	63
SF04_CSA 291-15_7.3.4_16	SF04-01-16	CSA 291-15	7.3.4	76
SF04_CSA N291-08_7.3.4_15	SF04-01-15	CSA N291-08	7.3.4	76

Category 4- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF05_CSA N288.2_6.4.1.1_16	SF05-12-16	CSA N288.2	6.4.1.1	136
SF05_CSA N288.2_6.5.1.1_16	SF05-12-16	CSA N288.2	6.5.1.1	137
SF08_SF8 RT_5.7_16	SF08-05-16	SF8 RT	5.7	144
SF12_NUREG-0700_Part_I_15	SF12-02-15	NUREG-0700	Part_I	191
SF12_NUREG-0700_Part_I_16	SF12-04-16	NUREG-0700	Part_I	191
SF12_NUREG-0700_Part_II_7_15	SF12-02-15	NUREG-0700	Part_II_7	196
SF12_NUREG-0700_Part_II_7_16	SF12-04-16	NUREG-0700	Part_II_7	196
SF12_NUREG-0700_Part_II_7_16	SF12-04-16	NUREG-0700	Part_II_7	197
SF12_NUREG-0700_Part_II_7_15	SF12-02-15	NUREG-0700	Part_II_7	197
SF12_SF12 RT_5.4_16	SF12-03-16	SF12 RT	5.4	199
SF12_SF12 RT_5.4_15	SF12-01-15	SF12 RT	5.4	199
SF12_SF12 RT_5.4_16	SF12-02-16	SF12 RT	5.4	200
SF13_IAEA GSR Part 7_5.49_16	SF13-02-16	IAEA GSR Part 7	5.49	201
SF13_IAEA GSR Part 7_5.52_16	SF13-02-16	IAEA GSR Part 7	5.52	202
SF13_IAEA GSR Part 7_5.53_16	SF13-02-16	IAEA GSR Part 7	5.53	203
SF13_IAEA GSR Part 7_5.57_16	SF13-02-16	IAEA GSR Part 7	5.57	204
SF13_IAEA GSR Part 7_5.60_16	SF13-02-16	IAEA GSR Part 7	5.60	205
SF13_IAEA GSR Part 7_6.11_16	SF13-02-16	IAEA GSR Part 7	6.11	206
SF13_CNSC REGDOC 2.10.1_2.2.4_15	SF13-01-15	CNSC REGDOC 2.10.1	2.2.4	208
SF13_CNSC REGDOC 2.10.1_2.2.6_16	SF13-01-16	CNSC REGDOC 2.10.1	2.2.6	209
SF13_CNSC REGDOC 2.10.1_2.2.6_15	SF13-01-15	CNSC REGDOC 2.10.1	2.2.6	209
SF13_CNSC REGDOC 2.10.1_2.2.8_16	SF13-01-16	CNSC REGDOC 2.10.1	2.2.8	210
SF13_CSA N1600-14_4.2.3_16	SF13-02-16	CSA N1600-14	4.2.3	216
SF13_CSA N1600-14_4.2.3_15	SF13-04-15	CSA N1600-14	4.2.3	216
SF13_CSA N1600-14_4.5.2_16	SF13-02-16	CSA N1600-14	4.5.2	217
SF13_CSA N1600-14_4.5.12_16	SF13-02-16	CSA N1600-14	4.5.12	217
SF13_CSA N1600-14_5.4_16	SF13-02-16	CSA N1600-14	5.4	217
SF13_CSA N1600-14_4.5.2_15	SF13-04-15	CSA N1600-14	4.5.2	217
SF13_CSA N1600-14_4.5.12_15	SF13-04-15	CSA N1600-14	4.5.12	217
SF13_CSA N1600-14_5.4_15	SF13-04-15	CSA N1600-14	5.4	217
SF13_SF13 RT 2015_7.4_15	SF13-01-15	SF13 RT 2015	7.4	219
SF13_SF13 RT 2016_7.4_16	SF13-01-16	SF13 RT 2016	7.4	219
SF13_SF13 RT 2016_5.1_16	SF13-01-16	SF13 RT 2016	5.1	221
SF13_SF13 RT 2016_5.1_16	SF13-01-16	SF13 RT 2016	5.1	222
SF13_SF13 RT 2016_5.3.1_16	SF13-03-16	SF13 RT 2016	5.3.1	223
SF13_SF13 RT 2016_5.3.1_16	SF13-03-16	SF13 RT 2016	5.3.1	224
SF13_SF13 RT 2016_5.3.1_16	SF13-03-16	SF13 RT 2016	5.3.1	225
SF13_SF13 RT 2016_5.3.1_16	SF13-03-16	SF13 RT 2016	5.3.1	226

Category 4- Micro-gap #	Macro-gap #	RCS/RT	Clause	Group #
SF13_SF13 RT 2016_5.3.2_16	SF13-03-16	SF13 RT 2016	5.3.2	227
SF13_SF13 RT 2016_5.3.3_16	SF13-03-16	SF13 RT 2016	5.3.3	228
SF13_SF13 RT 2016_5.3.3_16	SF13-03-16	SF13 RT 2016	5.3.3	229
SF13_SF13 RT 2016_7.1_16	SF13-01-16	SF13 RT 2016	7.1	230
SF13_SF13 RT 2016_7.2.1.1_16	SF13-01-16	SF13 RT 2016	7.2.1.1	231
SF15_SF15 RT_5.2.2_15	SF15-03-15	SF15 RT	5.2.2	239
SF15_SF15 RT_5.2.2_15	SF15-03-15	SF15 RT	5.2.2	239
SF15_SF15 RT_5.4.1_15	SF15-04-15	SF15 RT	5.4.1	240
SF15_WANO GL 2004-01-R1_VI.C2._15	SF15-03-15	WANO GL 2004-01-R1	VI.C2.	246
SF15_WANO GL 2004-01-R1_VI.C2._15	SF15-03-15	WANO GL 2004-01-R1	VI.C2.	247
SF15_WANO GL 2004-01-R1_VI.C2._15	SF15-03-15	WANO GL 2004-01-R1	VI.C2.	248
SF15_WANO GL 2004-01-R1_VI.C2._15	SF15-03-15	WANO GL 2004-01-R1	VI.C2.	249
SF15_WANO GL 2004-01-R1_VI.C2._15	SF15-03-15	WANO GL 2004-01-R1	VI.C2.	250
SF15_WANO GL 2004-01-R1_VI.C2._15	SF15-03-15	WANO GL 2004-01-R1	VI.C2.	251
SF14_CSA N288.3.4-13_8.5_15	SF14-01-15	CSA N288.3.4-13	8.5	257
SF14_CSA N288.3.4-13_ 8.9_15	SF14-01-15	CSA N288.3.4-13	8.9	258
SF14_CSA N288.3.4-13_10_15	SF14-01-15	CSA N288.3.4-13	10	259
SF14_CSA N288.3.4-13_11_15	SF14-01-15	CSA N288.3.4-13	11	260
SF14_CSA N288.3.4-13_12_15	SF14-01-15	CSA N288.3.4-13	12	261
SF14_CSA N288.3.4-13_13_15	SF14-01-15	CSA N288.3.4-13	13	262

Table 55: Micro-gaps with Safety Improvements Considered Necessary

Gap #	SF01_ANSI/NIRMA CM 1.0-2007_3.2_16
Document ID	ANSI/NIRMA CM 1.0-2007
Article/Clause	3.2
Requirement Assessed	Section 3 lists the criteria for the six areas included in successful implementation of configuration management for nuclear facilities.
Macro-Gap	SF01-24-16
Issue/Gap Description	As identified in Section 2.4 of the PSR Basis Document [26], a comprehensive review of the tracking of licence concessions granted to Bruce Power by the Regulator was conducted [27]. It was concluded that Bruce Power should establish a controlled, centralized and accessible company database of licence concessions to support design activities.

Gap #	SF01_CNSC REGDOC 2.5.2_8.13.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13.1 Design for radiation protection
Requirement Assessed	The shielding design shall prevent radiation levels in operating areas from exceeding the prescribed limits. This shall include provision of appropriate permanent layout and shielding of SSCs containing radioactive materials, and the use of temporary shielding for maintenance and inspection work. To minimize radiation exposure, the plant layout shall provide for efficient operation, inspection, maintenance, and replacement. In addition, the design shall limit the amount of activated material and its build-up. The design shall account for frequently occupied locations, and support the need for human access to locations and equipment. Access routes shall be shielded where needed. The design shall enable operator access for actions credited for post-accident conditions. Adequate protection shall be provided against exposure to radiation and radioactive contamination during DBAs and DECs for those parts of the facility to which access is required. Guidance Shielding should be designed based on the zone delineation described in section 8.13. The shielding design criteria (including the methodology for shield parameters and choice of shield material) should be provided. In establishing specifications for shielding, account should be taken of the buildup of radioactive materials over the lifetime of the NPP.
Macro-Gap	SF01-14-15
Issue/Gap Description	It is noted that although the criteria and rationale for radiation zone designations (for normal operations) are given in Part 2 of the Safety Report Section 12.3.3, the criteria and rationale are limited to what systems and qualitative probability of contamination there are in the area. Predicted dose rates or airborne radionuclides have not been explicitly considered. Therefore, this is assessed as a gap (Gap 2): There is no design documentation of the basis for station zoning for normal operations including consideration of the predicted dose rates or anticipated airborne radionuclides in the areas. Contamination levels are addressed in the definitions given in the Safety Report Section 12.3.3 and Appendix A of BP-RPP-00015.

Gap #	SF01_CNSC REGDOC 2.5.2_8.13.1_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13.1 Design for radiation protection
Requirement Assessed	The shielding design shall prevent radiation levels in operating areas from exceeding the prescribed limits. This shall include provision of appropriate permanent layout and shielding of SSCs containing radioactive materials, and the use of temporary shielding for maintenance and inspection work. To minimize radiation exposure, the plant layout shall provide for efficient operation, inspection, maintenance, and replacement. In addition, the design shall limit the amount of activated material and its build-up. The design shall account for frequently occupied locations, and support the need for human access to locations and equipment. Access routes shall be shielded where needed. The design shall enable operator access for actions credited for post-accident conditions. Adequate protection shall be provided against exposure to radiation and radioactive contamination during DBAs and DECs for those parts of the facility to which access is required. Guidance Shielding should be designed based on the zone delineation described in section 8.13. The shielding design criteria (including the methodology for shield parameters and choice of shield material) should be provided. In establishing specifications for shielding, account should be taken of the buildup of radioactive materials over the lifetime of the NPP.
Macro-Gap	SF01-14-15
Issue/Gap Description	The shielding design criteria and the methodology for shield parameters and choice of shield material are not sufficiently described in the design documentation. The buildup of radioactive materials over the lifetime of the NPP is not reflected in the shielding specifications as required in the guidance section; therefore it is assessed as a gap (Gap 1).

Gap #	SF01_CNSC REGDOC 2.5.2_8.13.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13.1 Design for radiation protection
Requirement Assessed	The shielding design shall prevent radiation levels in operating areas from exceeding the prescribed limits. This shall include provision of appropriate permanent layout and shielding of SSCs containing radioactive materials, and the use of temporary shielding for maintenance and inspection work. To minimize radiation exposure, the plant layout shall provide for efficient operation, inspection, maintenance, and replacement. In addition, the design shall limit the amount of activated material and its build-up. The design shall account for frequently occupied locations, and support the need for human access to locations and equipment. Access routes shall be shielded where needed. The design shall enable operator access for actions credited for post-accident conditions. Adequate protection shall be provided against exposure to radiation and radioactive contamination during DBAs and DECs for those parts of the facility to which access is required. Guidance Shielding should be designed based on the zone delineation described in section 8.13. The shielding design criteria (including the methodology for shield parameters and choice of shield material) should be provided. In establishing specifications for shielding, account should be taken of the buildup of radioactive materials over the lifetime of the NPP.
Macro-Gap	SF01-14-16
Issue/Gap Description	The shielding design criteria and the methodology for shield parameters and choice of shield material are not sufficiently described in the design documentation. The buildup of radioactive materials over the lifetime of the NPP is not reflected in the shielding specifications as required in the guidance section; therefore it is assessed as a gap (Gap 1).

Gap #	SF01_CNSC REGDOC 2.5.2_8.13.1_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13.1 Design for radiation protection
Requirement Assessed	The shielding design shall prevent radiation levels in operating areas from exceeding the prescribed limits. This shall include provision of appropriate permanent layout and shielding of SSCs containing radioactive materials, and the use of temporary shielding for maintenance and inspection work. To minimize radiation exposure, the plant layout shall provide for efficient operation, inspection, maintenance, and replacement. In addition, the design shall limit the amount of activated material and its build-up. The design shall account for frequently occupied locations, and support the need for human access to locations and equipment. Access routes shall be shielded where needed. The design shall enable operator access for actions credited for post-accident conditions. Adequate protection shall be provided against exposure to radiation and radioactive contamination during DBAs and DECs for those parts of the facility to which access is required. Guidance Shielding should be designed based on the zone delineation described in section 8.13. The shielding design criteria (including the methodology for shield parameters and choice of shield material) should be provided. In establishing specifications for shielding, account should be taken of the buildup of radioactive materials over the lifetime of the NPP.
Macro-Gap	SF01-14-16
Issue/Gap Description	It is noted that although the criteria and rationale for radiation zone designations (for normal operations) are given in Part 2 of the Safety Report Section 12.3.3, the criteria and rationale are limited to what systems and qualitative probability of contamination there are in the area. Predicted dose rates or airborne radionuclides have not been explicitly considered. Therefore, this is assessed as a gap (Gap 2): There is no design documentation of the basis for station zoning for normal operations including consideration of the predicted dose rates or anticipated airborne radionuclides in the areas. Contamination levels are addressed in the definitions given in the Safety Report Section 12.3.3 and Appendix A of BP-RPP-00015.

Gap #	SF01_CNSC REGDOC 2.5.2_8.13_15
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13 Radiation protection
Requirement Assessed	The design and layout of the plant shall make suitable provision to minimize exposure and contamination from all sources. This shall include the adequate design of SSCs to: 1. control access to the plant 2. minimize exposure during maintenance and inspection 3. provide shielding from direct and scattered radiation 4. provide ventilation and filtering to control airborne radioactive materials 5. limit the activation of corrosion products by proper specification of materials 6. minimize the spread of active material 7. monitor radiation levels 8. provide suitable decontamination facilities Guidance The NPP should be divided into zones based on predicted dose rates, radioactive contamination levels, concentration of airborne radionuclides, access requirements and specific requirements (such as the need to separate safety trains). The criteria and rationale for radiation zone designations – including zone boundaries for normal, refuelling and accident conditions – should be provided. These criteria should be used as the basis for the radiation shielding design. From a radiological protection perspective, careful assessment should be made of the access requirements for operation, inspection, maintenance, repair, replacement and decommissioning of equipment; these considerations should be incorporated into the design. The design should also provide lay down space for special tools and ease for servicing activities. The design should also have features such as platforms or walkways, stairs, or ladders that permit prompt accessibility for servicing or inspection of components located in higher radiation zones. The use of remote technology for maintenance and surveillance in high radiation areas should be considered and incorporated. Preference should be given to the use of appropriate engineering controls and design

	features over process or administrative controls. Reliable equipment that requires minimum surveillance, maintenance, testing and calibration should be chosen. Operating experience should be reflected in the criteria and rationale provided in the design.
Macro-Gap	SF01-14-15
Issue/Gap Description	The criteria and rationale for radiation zone designations – including zone boundaries for accident conditions are not provided in the design documentation as suggested in guidance. The criteria and rationale seem, however, to be limited to what systems and qualitative probability of contamination there are in the area. There doesn't seem to be any consideration of predicted dose rates or airborne radionuclides. There is no documentation of the basis for station zoning for normal operations including consideration of the predicted dose rates or anticipated airborne radionuclides in the areas. Zone boundaries are not provided in the design. Therefore, it is assessed as a gap (Gap). It is recognized that such expectations are more relevant to new reactor designs.

Gap #	SF01_CNSC REGDOC 2.5.2_8.13_16
Document ID	CNSC REGDOC 2.5.2
Article/Clause	8.13 Radiation protection
Requirement Assessed	The design and layout of the plant shall make suitable provision to minimize exposure and contamination from all sources. This shall include the adequate design of SSCs to: 1. control access to the plant 2. minimize exposure during maintenance and inspection 3. provide shielding from direct and scattered radiation 4. provide ventilation and filtering to control airborne radioactive materials 5. limit the activation of corrosion products by proper specification of materials 6. minimize the spread of active material 7. monitor radiation levels 8. provide suitable decontamination facilities Guidance The NPP should be divided into zones based on predicted dose rates, radioactive contamination levels, concentration of airborne radionuclides, access requirements and specific requirements (such as the need to separate safety trains). The criteria and rationale for radiation zone designations – including zone boundaries for normal, refuelling and accident conditions – should be provided. These criteria should be used as the basis for the radiation shielding design. From a radiological protection perspective, careful assessment should be made of the access requirements for operation, inspection, maintenance, repair, replacement and decommissioning of equipment; these considerations should be incorporated into the design. The design should also provide lay down space for special tools and ease for servicing activities. The design should also have features such as platforms or walkways, stairs, or ladders that permit prompt accessibility for servicing or inspection of components located in higher radiation zones. The use of remote technology for maintenance and surveillance in high radiation areas should be considered and incorporated. Preference should be given to the use of appropriate engineering controls and design

	features over process or administrative controls. Reliable equipment that requires minimum surveillance, maintenance, testing and calibration should be chosen. Operating experience should be reflected in the criteria and rationale provided in the design.
Macro-Gap	SF01-14-16
Issue/Gap Description	The criteria and rationale for radiation zone designations – including zone boundaries for accident conditions are not provided in the design documentation as suggested in guidance. The criteria and rationale seem, however, to be limited to what systems and qualitative probability of contamination there are in the area. There does not seem to be any consideration of predicted dose rates or airborne radionuclides. There is no documentation of the basis for station zoning for normal operations including consideration of the predicted dose rates or anticipated airborne radionuclides in the areas. Zone boundaries are not provided in the design. Therefore, it is assessed as a gap (Gap). It is recognized that such expectations are more relevant to new reactor designs.

Gap #	SF01_CSA N289.1_3.1_16
Document ID	CSA N289.1
Article/Clause	3.1
Requirement Assessed	It is recommended that the governing procedure (DPT-PDE-00017) and its implementing documents (NK29-DG-03650-002 [140]) be updated to reflect the latest requirements of clause 3.1 from CSA N289.1.
Macro-Gap	SF01-16-16
Issue/Gap Description	As described in section 5.3.3 of SF1 report: The requirements in the latest revision are generally more detailed and provide an update to reflect current practices. This is assessed in Safety Factor 3, which recommends the governing procedure (DPT-PDE-00017 [142]) and its implementing documents (NK29-DG-03650-002 [143]) be updated to reflect the latest requirements of clause 3.1 from CSA N289.1 (i.e., the 10-4 requirement for the definition of the Design Basis Earthquake (DBE)), including the 2014 update. The reporting and recording requirements for earthquake events and the more recent site investigations documented in the Probabilistic Seismic Hazard Assessment performed in 2011 [157] are not reflected in the seismic implementing procedures.

Gap #	SF01_CSA N290.0-11_4.5-4.8_15
Document ID	CSA N290.0-11
Article/Clause	4.5-4.8
Requirement Assessed	Present the requirements related to reliability, separation and independence, single failure criteria application and fail-safe design concept.
Macro-Gap	SF01-05-15
Issue/Gap Description	Gap against clause 4.5.2.1: The probability of failure on demand from all causes for some safety systems has not been consistently lower than 1E-3.

Gap #	SF01_CSA N290.1_4.2.1.1_15
Document ID	CSA N290.1
Article/Clause	4.2.1.1 Probability of failure to shutdown on demand
Requirement Assessed	The reliability evaluation shall demonstrate that the reliability of the shutdown function from all credited means is such that the cumulative probability of failure to shutdown on demand can be shown to meet its requirement. The contribution of all sequences, involving failure to shutdown, to the large release frequency shall be less than the target stated in regulatory requirements. Notes: 1) General requirements on reliability and reliability analysis for safety systems can be found in CSA N290.0, Clause 4.5. 2) The probability of an SDS failure on demand for existing CANDU plants is typically lower than 1E-3. 3) CNSC RD/GD-98 requires a licensee who constructs or operates an NPP to develop and implement a reliability program that assures that the systems important to safety can and will meet their defined design and performance specifications at acceptable levels of reliability throughout the lifetime of the NPP.
Macro-Gap	SF01-05-15
Issue/Gap Description	The reliability targets are specified in the design manuals. The annual reliability reports show that the probability of failure on demand from all causes for some safety systems has not been consistently lower than 1E-3. The current PRA results indicate that the contribution of all sequences, involving failure to shut down, to the large release frequency is about 2.3x1E-7, which is higher than REGDOC-2.5.2 proposed target of 1E-7 (Gap). See Safety Factor 6 for more details.

Gap #	SF01_CSA N290.1_4.7.2_15
Document ID	CSA N290.1
Article/Clause	4.7.2 Qualification for electromagnetic and mechanical disturbances
Requirement Assessed	To achieve high signal sensitivity and avoid spurious SDS actuation, the SSCs employed in the design of the SDS shall be qualified for electromagnetic noise disturbances (conducted and radiated, continuous and transient) and mechanical vibrations from normally operating plant equipment. Qualification tests shall be specified and performed to provide assurance that electromagnetic and mechanical disturbances cannot render the SDS ineffective.
Macro-Gap	SF01-12-15
Issue/Gap Description	Bruce A design documentation (e.g., Safety Report, Bruce Power DM's, etc.) does not explicitly state the SSCs employed are qualified for electromagnetic noise disturbances and mechanical vibrations (Gap 1). Qualification against electromagnetic susceptibility for the installed equipment cannot be confirmed. As such, the requirement for the clause is deemed not in compliance. Bruce Power is implementing compensatory measures to avoid spurious trips. For example, all instrument rooms (R-317, 316 and 211) at Bruce A are designed as 'radio-free' zones. Roll-outs to all control maintenance personnel and MCR operations staff have been completed to enforce the expectations on radio use in or around the instrument rooms, the vertical reactivity deck, gantry crane movement activities or any maintenance that takes place on SDS equipment in the vault. It is common practice for Bruce Power to request EMI/RF qualification for all new I&C components, which is typically documented in the Technical Specifications.

Gap #	SF01_CSA N290.1_4.7.2_16
Document ID	CSA N290.1
Article/Clause	4.7.2 Qualification for electromagnetic and mechanical disturbances
Requirement Assessed	To achieve high signal sensitivity and avoid spurious SDS actuation, the SSCs employed in the design of the SDS shall be qualified for electromagnetic noise disturbances (conducted and radiated, continuous and transient) and mechanical vibrations from normally operating plant equipment. Qualification tests shall be specified and performed to provide assurance that electromagnetic and mechanical disturbances cannot render the SDS ineffective.
Macro-Gap	SF01-12-16
Issue/Gap Description	Bruce B design documentation (e.g., Safety Report, Bruce Power DM's, etc.) does not explicitly state the SSCs employed are qualified for electromagnetic noise disturbances and mechanical vibrations (Gap 1).

Gap #	SF01_CSA N290.11-13_5.2.2.4_15
Document ID	CSA N290.11-13
Article/Clause	5.2.2.4
Requirement Assessed	Staff credited with performing contingency activities to support the heat sink are required not to be credited with availability for other activities.
Macro-Gap	SF01-07-15
Issue/Gap Description	The operating documentation does not explicitly reflect the requirement that the staff credited with performing contingency activities to support the heat sink not to be credited with availability for other activities.

Gap #	SF01_CSA N290.11-13_5.2.2.4_16
Document ID	CSA N290.11-13
Article/Clause	5.2.2.4
Requirement Assessed	Staff credited with performing contingency activities to support the heat sink are required not to be credited with availability for other activities.
Macro-Gap	SF01-07-16
Issue/Gap Description	Lack of demonstration that staff credited with performing contingency activities to support the heat sink are not credited with availability for other activities.

Gap #	SF03_CSA N289.1_5_16
Document ID	CSA N289.1
Article/Clause	5
Requirement Assessed	General Seismic Requirements
Macro-Gap	SF03-05-16
Issue/Gap Description	The governing procedure (DPT-PDE-00017 [64]) and its implementing documents (NK29-DG-03650-002 [65]) have not been updated to reflect the latest requirements of CSA N289.1 (i.e., the 1E-4 requirement for the definition of the DBE), including the 2014 update, nor the results of the Probabilistic Seismic Hazard Assessment done in 2011.

Gap #	SF04_CSA 291-15_7.3.4_16
Document ID	CSA 291-15
Article/Clause	7.3.4 Abnormal/environmental conditions
Requirement Assessed	Following any abnormal/environmental condition, all structural components shall be subjected to a visual inspection and other methods of examination, as required, to evaluate the integrity of the structure.
Macro-Gap	SF04-01-16
Issue/Gap Description	NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures does not describe inspection requirements following an abnormal/environmental condition.

Gap #	SF04_CSA N291-08_7.3.4_15
Document ID	CSA N291-08
Article/Clause	7.3.4 Abnormal/environmental conditions
Requirement Assessed	Following any abnormal/environmental condition, all structural components shall be subjected to a visual inspection and other methods of examination, as required, to evaluate the integrity of the structure.
Macro-Gap	SF04-01-15
Issue/Gap Description	NK21-PIP-20000-00001, "CSA N291 In-Service Inspection Program for Bruce NGS A Safety Related Structures" does not describe inspection requirements following an abnormal/environmental condition. Consideration should be given to revising NK21-PIP-20000-00001 to include inspection requirements following an abnormal/environmental condition.

Gap #	SF05_CSA N288.2_6.4.1.1_16
Document ID	CSA N288.2
Article/Clause	6.4.1.1
Requirement Assessed	The use of specialized models shall be considered wherever the release occurs over rugged terrain, at a coastline, or in a region of large land-use variations.
Macro-Gap	SF05-12-16
Issue/Gap Description	Since the site is located on the shore of Lake Huron, a specialized shoreline dispersion model may be needed.

Gap #	SF05_CSA N288.2_6.5.1.1_16
Document ID	CSA N288.2
Article/Clause	6.5.1.1
Requirement Assessed	Justification of the chosen model should be provided for each application. In particular, the model shall produce results that meet the goals of the safety assessment.
Macro-Gap	SF05-12-16
Issue/Gap Description	Bruce B Safety Report Part 3 does not include justification for the atmospheric dispersion model selected for the dose calculations

Gap #	SF08_SF8 RT 2015_5.8_15
Document ID	SF8 RT 2015
Article/Clause	5.8
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: f) Compliance with regulatory requirements
Macro-Gap	SF08-09-15
Issue/Gap Description	Standby Class III Power System predicted unavailability targets exceeded in 2012 and 2013 due to an inconsistency between the modelling and plant operation. This requires correction action to reduce the unavailability.

Gap #	SF08_SF8 RT_5.7_16
Document ID	SF8 RT
Article/Clause	5.7
Requirement Assessed	The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including: e) Modifications, either temporary or permanent, to SSCs important to safety
Macro-Gap	SF08-05-16
Issue/Gap Description	It is unclear where postulated initiating events involving hazard analyses of this nature are documented to ensure the adequacy of protection of the NPP against internal and external hazards are registered as part of the analysis and assessments of records. Presently these safety assessments tend to be in various documents (e.g., Seismic, Pipe-whip and Fire [255] [258]) so it would be useful to provide an integrating document to confirm completeness, to ensure the hazard assessments remain current as knowledge is improved and modifications are made to the SSCs, and the integration and overlap with Deterministic Safety Analysis and Probabilistic Safety Assessments are well known.

Gap #	SF12_NUREG-0700_Part_I_15
Document ID	NUREG-0700
Article/Clause	Part_I
Requirement Assessed	Part I of NUREG-0700 provides guidelines for the basic HSI elements: information display, user interface interaction and management and controls
Macro-Gap	SF12-02-15
Issue/Gap Description	No guidelines identified for CRT based displays for Bruce A, even though such technologies are in use for the display of information. Therefore, it is not clear for engineering changes that are applied to such systems, what guidelines are to be used and the extent which the review conducted against NUREG-0700 applies to Bruce A. This gap also applies to Part 2, Soft Control Systems.

Gap #	SF12_NUREG-0700_Part_I_16
Document ID	NUREG-0700
Article/Clause	Part_I
Requirement Assessed	Part I of NUREG-0700 provides guidelines for the basic HSI elements: information display, user interface interaction and management and controls
Macro-Gap	SF12-04-16
Issue/Gap Description	No guidelines identified for CRT based displays for Bruce A, even though such technologies are in use for the display of information. Therefore, it is not clear for engineering changes that are applied to such systems, what guidelines are to be used and the extent to which the review conducted against NUREG-0700 applies to Bruce A. This gap also applies to Part 2, Soft Control Systems.

Gap #	SF12_NUREG-0700_Part_II_7_15
Document ID	NUREG-0700
Article/Clause	Part_II_7
Requirement Assessed	Guidelines for reviewing Soft Control Systems
Macro-Gap	SF12-02-15
Issue/Gap Description	B-DG-06700-00004, Human Factors Design Guide for Computer Interfaces does not reference NUREG-0700, nor does it reference any other guideline, standard, or code.

Gap #	SF12_NUREG-0700_Part_II_7_15
Document ID	NUREG-0700
Article/Clause	Part_II_7
Requirement Assessed	Guidelines for reviewing Soft Control Systems
Macro-Gap	SF12-02-15
Issue/Gap Description	As a result of B-DG-06700-00004, Human Factors Design Guide for Computer Interfaces not referencing any guideline, standard, or code for source of the guidance, it could not be established whether soft control systems are reviewed against NUREG-0700 or any other modern guideline or standard.

Gap #	SF12_NUREG-0700_Part_II_7_16
Document ID	NUREG-0700
Article/Clause	Part_II_7
Requirement Assessed	Guidelines for reviewing Soft Control Systems
Macro-Gap	SF12-04-16
Issue/Gap Description	As a result of B-DG-06700-00004, Human Factors Design Guide for Computer Interfaces not referencing any guideline, standard, or code for source of the guidance, it could not be established whether soft control systems are reviewed against NUREG-0700 or any other modern guideline or standard.

Gap #	SF12_NUREG-0700_Part_II_7_16
Document ID	NUREG-0700
Article/Clause	Part_II_7
Requirement Assessed	Guidelines for reviewing Soft Control Systems
Macro-Gap	SF12-04-16
Issue/Gap Description	B-DG-06700-00004, Human Factors Design Guide for Computer Interfaces does not reference NUREG-0700, nor does it reference any other guideline, standard, or code.

Gap #	SF12_SF12 RT_5.4_15
Document ID	SF12 RT
Article/Clause	5.4
Requirement Assessed	This review task includes review of the Bruce safety analysis programs to ensure that assumptions and claims made about Operator actions under accident conditions have been assessed and confirmed valid.
Macro-Gap	SF12-01-15
Issue/Gap Description	It is possible that not all operator actions under accident conditions were assessed and validated

Gap #	SF12_SF12 RT_5.4_16
Document ID	SF12 RT
Article/Clause	5.4
Requirement Assessed	This review task includes review of the Bruce safety analysis programs to ensure that assumptions and claims made about Operator actions under accident conditions have been assessed and confirmed valid.
Macro-Gap	SF12-02-16
Issue/Gap Description	Lack of input from training exercises, particularly those modeling accident conditions, to safety analyses to validate assumptions.

Gap #	SF12_SF12 RT_5.4_16
Document ID	SF12 RT
Article/Clause	5.4
Requirement Assessed	This review task includes review of the Bruce safety analysis programs to ensure that assumptions and claims made about Operator actions under accident conditions have been assessed and confirmed valid.
Macro-Gap	SF12-03-16
Issue/Gap Description	A review of Bruce Power documentation could not confirm that all operator actions under accident conditions have been assessed and confirmed valid. While it is clear that all credited human actions, as noted in the Bruce B Risk Assessment Report and included in AIMs were validated, it is not clear whether human actions identified in the Bruce B Safety Report were a part of the credited human actions validated.

Gap #	SF13_IAEA GSR Part 7_5.49_16
Document ID	IAEA GSR Part 7
Article/Clause	5.49
Requirement Assessed	Arrangements shall be made to ensure that emergency workers are, to the extent practicable, designated in advance and are fit for the intended duty. These arrangements shall include health surveillance for emergency workers for the purpose of assessing their initial fitness and continuing fitness for their intended duties.
Macro-Gap	SF13-02-16
Issue/Gap Description	Fitness for duty is currently addressed through various station programs for all staff. However, these are considered increased expectation. It is also noted that Draft REGDOC-2.2.4, Human Performance Management - Fitness for Duty, was issued by the CNSC in November 2015.

Gap #	SF13_IAEA GSR Part 7_5.52_16
Document ID	IAEA GSR Part 7
Article/Clause	5.52
Requirement Assessed	The operating organization and response organizations shall ensure that arrangements are in place for the protection of emergency workers and protection of helpers in an emergency for the range of anticipated hazardous conditions in which they might have to perform response functions. These arrangements, as a minimum, shall include: (a) Training those emergency workers designated as such in advance; (b) Providing emergency workers not designated in advance and helpers in an emergency immediately before the conduct of their specified duties with instructions on how to perform the duties under emergency conditions ('just in time' training); (c) Managing, controlling and recording the doses received; (d) Provision of appropriate specialized protective equipment and monitoring equipment; (e) Provision of iodine thyroid blocking, as appropriate, if exposure due to radioactive iodine is possible; (f) Obtaining informed consent to perform specified duties, when appropriate; (g) Medical examination, longer term medical actions and psychological counselling, as appropriate.
Macro-Gap	SF13-02-16
Issue/Gap Description	As part of the arrangements for the protection of emergency workers and protection of helpers in an emergency for the range of anticipated hazardous conditions in which they might have to perform response functions does not include: Medical examination, longer term medical actions and psychological counselling, as appropriate.

Gap #	SF13_IAEA GSR Part 7_5.53_16
Document ID	IAEA GSR Part 7
Article/Clause	5.53
Requirement Assessed	The operating organization and response organizations shall ensure that all practicable means are used to minimize exposures of emergency workers and helpers in an emergency in the response to a nuclear or radiological emergency (see para. I.2 of Appendix I), and to optimize their protection.
Macro-Gap	SF13-02-16
Issue/Gap Description	Clause 5.53 states that; "The operating organization and response organizations shall ensure that all practicable means are used to minimize exposures of emergency workers and helpers in an emergency in the response to a nuclear or radiological emergency (see para. I.2 of Appendix I), and to optimize their protection." Currently there is no means to demonstrate "all practical means are used… to minimize exposures... and .....optimize their protection"

Gap #	SF13_IAEA GSR Part 7_5.57_16
Document ID	IAEA GSR Part 7
Article/Clause	5.57
Requirement Assessed	The operating organization and response organizations shall ensure that emergency workers who undertake emergency response actions in which doses received might exceed an effective dose of 50 mSv do so voluntarily; that they have been clearly and comprehensively informed in advance of associated health risks as well as of available protective measures; and that they are, to the extent possible, trained in the actions that they might be required to take. Emergency workers not designated as such in advance shall not be the first emergency workers chosen for taking actions that could result in their doses exceeding the guidance values of dose for lifesaving actions, as given in Appendix I. Helpers in an emergency shall not be allowed to take actions that could result in their receiving doses in excess of an effective dose of 50 mSv.
Macro-Gap	SF13-02-16
Issue/Gap Description	Currently, there are no clear criteria for authorizing exceeding dose limits in the Emergency Preparedness Program and supporting documentation.

Gap #	SF13_IAEA GSR Part 7_5.60_16
Document ID	IAEA GSR Part 7
Article/Clause	5.60
Requirement Assessed	Emergency workers who receive doses in a response to a nuclear or radiological emergency shall normally not be precluded from incurring further occupational exposure. However, qualified medical advice shall be obtained before any further occupational exposure occurs if an emergency worker has received an effective dose exceeding 200 mSv, or at the request of the emergency worker.
Macro-Gap	SF13-02-16
Issue/Gap Description	Obtaining qualified medical advice is a change from IAEA-GS-R-2, and is currently not included in the Emergency Preparedness Program and supporting documentation.

Gap #	SF13_IAEA GSR Part 7_6.11_16
Document ID	IAEA GSR Part 7
Article/Clause	6.11
Requirement Assessed	For a site where multiple facilities in category I or II are co-located, an appropriate number of suitably qualified personnel shall be available to manage an emergency response at all facilities if each of the facilities is under emergency conditions simultaneously (see para. 5.4).
Macro-Gap	SF13-02-16
Issue/Gap Description	The scope of the multi-unit emergency staffing requirement and minimum staffing complement is unresolved (see NK29-CORR-00531-12798).

Gap #	SF13_CNSC REGDOC 2.10.1_2.2.4_15
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.2.4 Interface and support for offsite response organizations
Requirement Assessed	All licensees shall: In accordance with ER plans and procedures: 1. establish plans and procedures to coordinate response activities with appropriate offsite organizations, in the event of an emergency with offsite implications 2. formally document any arrangements or agreements with other organizations or personnel 3. ensure that agreed-upon resources, and the quantity of these resources required to respond to offsite conditions, are available when needed 4. cooperate with and assist offsite organizations with their response activities to address offsite impacts; provide expertise and resources (personnel, emergency response equipment, and material) in support of offsite authorities during an emergency; and define the quantity of available resources within their ER plan 5. promptly and regularly provide recommendations to offsite authorities when protective action is required and inform the CNSC 6. establish what data is required and at what frequency, and make provisions to have nuclear facility data, and any other pertinent information that is determined as relevant to the emergency response, regularly transmitted to offsite authorities and the CNSC Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW. These licensees shall: 1. incorporate the provincial or territorial emergency planning zone that is being used for plume exposure and ingestion pathways; the provincial or territorial plans shall be directly referenced 2. collaborate with the municipal or regional authorities to develop and maintain public evacuation time estimates based on current census data, and future population growth projections on a per-decade estimation until end of life of the facility 3. have, at all times, a designated onsite person with the authority and responsibility to categorize a nuclear emergency and to perform the following promptly and without consultation, upon categorization of the emergency: a. initiate an appropriate onsite response b. notify the appropriate offsite authorities c. provide sufficient information for an effective offsite response 4. provide the designated person with a suitable means of alerting onsite response personnel and notifying the offsite notification point 5. for NPPs, ensure there is a designated person onsite at all times with the authority for venting

	6. for NPPs, ensure that offsite authorities and the CNSC are consulted before undertaking any venting activity, unless venting must be performed in an urgent manner to protect the structural integrity of containment; in such a case, every effort shall be made to inform the offsite authorities and the CNSC as early as possible 7. include, in each report to the CNSC and offsite authorities, estimates of when venting will be required 8. notify the province and the CNSC of all abnormal incidents as described in section 2.2.2 Guidance Guidance for all licensees Licensees should identify the jurisdictions, organizations or persons that could be formally involved in emergency preparedness and response activities pertaining to facility emergencies with offsite impacts, and then develop mutual aid and community agreements where appropriate. During an emergency it is critical to have an onsite person with the required authority to order emergency venting if required. However, this authority can be delegated if it is impractical to have a senior emergency officer onsite at all times. The ER plan should also define a clear and concise strategy for communications between onsite and offsite organizations. All communications, including event data and the decisions made throughout the emergency response, should be documented and recorded. While the licensee is required to provide recommendations to offsite authorities, it is at the discretion of the authorities to accept, reject or modify recommendations. The nuclear emergency response plans for offsite response organizations (those of provinces and municipalities as well as firefighters, emergency medical services personnel and police) should be included with licence application documents for licence renewal and new applications.
Macro-Gap	SF13-01-15
Issue/Gap Description	General improvements/revisions to the Emergency Measures Program, the BPNERP, and implementing documents is required, which includes: • providing recommendations to off site authorities;

Gap #	SF13_CNSC REGDOC 2.10.1_2.2.6_15
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.2.6 Emergency response facilities and equipment
Requirement Assessed	All licensees shall In accordance with ER plans and procedures: 1. identify an onsite emergency response facility or designated area to be used as a response location 2. identify essential emergency response equipment, and describe how its operation and effectiveness during emergencies are assured; essential emergency response equipment includes equipment required to detect and assess hazards, and communicate response activities 3. identify and have emergency response equipment and materials that are operational and available in sufficient quantities for an extended multi-shift response; they shall also be readily accessible during emergency conditions Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW. These licensees shall: 4. have an emergency response facility (ERF) located onsite, but outside of the protected area; if this cannot be achieved, describe security arrangements to prevent nuisance actors from interfering with emergency response, and provisions for alternate means of communication in the event of a total communications blackout 5. have an emergency response facility located offsite and outside of the plume exposure planning zone 6. ensure that the emergency response facilities will ensure the health and safety of workers in the ERF and ensure the continuity of operations for all emergency situations that cannot be practically eliminated (if this cannot be achieved, then have backup facility with similar capability for each of the onsite and offsite such that the backup facility is unlikely to be effected by an event that would disable the primary; in addition, activation or transfer of operations to the backup facility must be done without disruption to the response operations) 7. provide a workspace with computer, internet access and telephone for a CNSC representative in each ERF; in addition, the CNSC shall be granted access to install an antenna for a satellite phone at each ERF 8. ensure all emergency response facilities have the capacity and capability of sustaining emergency response for a minimum of 72 hours without offsite support 9. ensure the design and layout of emergency response facilities are able to support the emergency response 10. ensure emergency response facilities have provisions in place to provide nuclear facility data

	11. pre-arrange memoranda of understanding and/or other priority services agreements required to keep ERFs functional over prolonged periods, and ensure such agreements are documented and either referenced or attached to the ER plan 12. determine and implement methods for communicating with onsite personnel and offsite authorities, including the implementation of at least two levels of backup communications systems; licensee communication links must be compatible with the licensee, province or territory, and the CNSC Guidance Guidance for all licensees Licensees should describe the emergency response services, equipment, supplies and facilities that would be available during emergencies, including, but not limited to the following: • administration facilities • technical support centres • control facilities • personnel and public assembly areas • emergency operations coordination centre • centre to integrate onsite activities with offsite programs • first aid and/or medical facilities • laboratory services (fixed or mobile) • decontamination facility • backup power capable of sustaining emergency power to emergency response facilities for a minimum of 72 hours • reference materials, such as current and approved versions of charts, maps, plans, drawings, diagrams, specifications and procedures • essential safety equipment, PPE and other appropriate supplies, such as food and water for a minimum of 72 hours • administrative aids, such as status boards and reference materials • fixed or portable instruments or equipment, as required, to detect, measure, monitor, survey, analyze, record, process, treat, transport, warn, announce, communicate, or assess Additional guidance for licensees of reactor facilities with a thermal capacity greater than 10 MW The CNSC workspace should have appropriate resources (such as computers, information access, internet access and satellite phones) to enable CNSC representatives to perform their functions adequately. The preferred means of ensuring the protection of workers and the continuation of operation is to have hardened facilities within the primary zone that have: • radiological protection/shielding

	• adequate ventilation, • contamination control • the ability to withstand design-basis event hazards, such as wind, tornado, snow or ice
Macro-Gap	SF13-01-15
Issue/Gap Description	General improvements/revisions to the Emergency Measures Program, the BPNERP, and implementing documents is required, which includes: • ensuring security arrangements at off-site centres;

Gap #	SF13_CNSC REGDOC 2.10.1_2.2.6_16
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.2.6 Emergency response facilities and equipment
Requirement Assessed	All licensees shall In accordance with ER plans and procedures: 1. identify an onsite emergency response facility or designated area to be used as a response location 2. identify essential emergency response equipment, and describe how its operation and effectiveness during emergencies are assured; essential emergency response equipment includes equipment required to detect and assess hazards, and communicate response activities 3. identify and have emergency response equipment and materials that are operational and available in sufficient quantities for an extended multi-shift response; they shall also be readily accessible during emergency conditions Additional requirements for licensees of reactor facilities with a thermal capacity greater than 10 MW. These licensees shall: 4. have an emergency response facility (ERF) located onsite, but outside of the protected area; if this cannot be achieved, describe security arrangements to prevent nuisance actors from interfering with emergency response, and provisions for alternate means of communication in the event of a total communications blackout 5. have an emergency response facility located offsite and outside of the plume exposure planning zone 6. ensure that the emergency response facilities will ensure the health and safety of workers in the ERF and ensure the continuity of operations for all emergency situations that cannot be practically eliminated (if this cannot be achieved, then have backup facility with similar capability for each of the onsite and offsite such that the backup facility is unlikely to be effected by an event that would disable the primary; in addition, activation or transfer of operations to the backup facility must be done without disruption to the response operations) 7. provide a workspace with computer, internet access and telephone for a CNSC representative in each ERF; in addition, the CNSC shall be granted access to install an antenna for a satellite phone at each ERF 8. ensure all emergency response facilities have the capacity and capability of sustaining emergency response for a minimum of 72 hours without offsite support 9. ensure the design and layout of emergency response facilities are able to support the emergency response 10. ensure emergency response facilities have provisions in place to provide nuclear facility data

	11. pre-arrange memoranda of understanding and/or other priority services agreements required to keep ERFs functional over prolonged periods, and ensure such agreements are documented and either referenced or attached to the ER plan 12. determine and implement methods for communicating with onsite personnel and offsite authorities, including the implementation of at least two levels of backup communications systems; licensee communication links must be compatible with the licensee, province or territory, and the CNSC Guidance Guidance for all licensees Licensees should describe the emergency response services, equipment, supplies and facilities that would be available during emergencies, including, but not limited to the following: • administration facilities • technical support centres • control facilities • personnel and public assembly areas • emergency operations coordination centre • centre to integrate onsite activities with offsite programs • first aid and/or medical facilities • laboratory services (fixed or mobile) • decontamination facility • backup power capable of sustaining emergency power to emergency response facilities for a minimum of 72 hours • reference materials, such as current and approved versions of charts, maps, plans, drawings, diagrams, specifications and procedures • essential safety equipment, PPE and other appropriate supplies, such as food and water for a minimum of 72 hours • administrative aids, such as status boards and reference materials • fixed or portable instruments or equipment, as required, to detect, measure, monitor, survey, analyze, record, process, treat, transport, warn, announce, communicate, or assess Additional guidance for licensees of reactor facilities with a thermal capacity greater than 10 MW The CNSC workspace should have appropriate resources (such as computers, information access, internet access and satellite phones) to enable CNSC representatives to perform their functions adequately. The preferred means of ensuring the protection of workers and the continuation of operation is to have hardened facilities within the primary zone that have: • radiological protection/shielding

	• adequate ventilation, • contamination control • the ability to withstand design-basis event hazards, such as wind, tornado, snow or ice
Macro-Gap	SF13-01-16
Issue/Gap Description	While informal security arrangements are in place at the Emergency Management Center (EMC) and back-up EMCs, these need to be formalized in BP-PLAN-00001 and implementing procedures.

Gap #	SF13_CNSC REGDOC 2.10.1_2.2.8_16
Document ID	CNSC REGDOC 2.10.1
Article/Clause	2.2.8 Recovery
Requirement Assessed	All licensees shall: In accordance with ER plans and procedures: 1. describe the process to transition from emergency response to recovery after the termination of an emergency, including the requirements to establish a recovery organization and to develop a recovery plan 2. identify, in the recovery plan, the positions/titles, authorities and responsibilities of the individuals who will fill key positions in the recovery organization; this organization shall also include technical personnel with responsibilities to develop, evaluate and direct recovery and reentry operations Guidance Guidance for all licensees A conceptual and strategic recovery plan should be prepared in advance. This can act as the basis for developing the recovery plan after the event has occurred and the emergency phase is complete. The recovery plan should: • identify and describe the resources (personnel, facilities and emergency response equipment) that are to be available for recovery purposes • describe how personnel will be protected when assessing or implementing the recovery program (e.g., personnel protection measures for entry into hazardous areas) • provide for post-accident assessments of the causes, details, impacts and/or consequences of the events • ensure all recovery efforts operate in accordance with the licensee's operating licence requirements Once the emergency phase of an emergency response has ended, workers undertaking recovery operations (such as repairs to plant and buildings, waste disposal or decontamination of the site and surrounding area) are subject to the occupational dose limits listed in the CNSC's Radiation Protection Regulations.
Macro-Gap	SF13-01-16
Issue/Gap Description	The BPNERP does not specify the process for developing recovery plans. However, per BP-PROG-08.01, recovery plans are identified through BP-PROC-00317, Crisis Management, and the use of business continuity procedures, with oversight provided by the Crisis Management Team. In

	accordance with this procedure, each business group is responsible for developing and maintaining their own recovery procedures. While the basic structure for recovery plans is in place, it is considered that the intent of REGDOC-2.10.1 is not fully met. Per NK29-CORR-00531-12566, a transition plan for full compliance with REGDOC-2.10.1 by August 31, 2018 is in place. A test of the concept of recovery planning is part of Exercise Huron Resolve in the fall of 2016. Lessons learned from this exercise will be incorporated into revised recovery plan governance. This is considered to be a gap.

Gap #	SF13_CSA N1600-14_4.2.3_15
Document ID	CSA N1600-14
Article/Clause	4.2.3 Impact analysis
Requirement Assessed	
Macro-Gap	SF13-04-15
Issue/Gap Description	Addressing the additional requirements in CSA N1600-14. There are a number of detailed additional requirements in CSA N1600-14 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: • an evaluation of losing critical functions which might impact the ability to respond and recover from an emergency. .... Given that CSA N1600-14 is likely to be substantially revised in the short term, a phased approach should be taken to its detailed review for elements that need to be addressed by Bruce Power.

Gap #	SF13_CSA N1600-14_4.2.3_16
Document ID	CSA N1600-14
Article/Clause	4.2.3
Requirement Assessed	An evaluation of losing critical functions which might impact the ability to respond and recover from an emergency with the goal being to ensure continuity of the critical functions.
Macro-Gap	SF13-02-16
Issue/Gap Description	CSA N1600 requires an evaluation of losing critical functions which might impact the ability to respond and recover from an emergency with the goal being to ensure continuity of the critical functions (critical functions cover more than equipment). There is no equivalent requirement in CNSC REGDOC-2.10.1. The latter requires identification of essential emergency response equipment, and a description of how their operation and effectiveness in an emergency are assured. In addition, CNSC REGDOC 2.3.2 requires demonstration with reasonable assurance that equipment and instrumentation used in severe accident management will survive and perform their required function.

Gap #	SF13_CSA N1600-14_4.5.12_15
Document ID	CSA N1600-14
Article/Clause	4.5.12 Deviation from the nuclear emergency response plan
Requirement Assessed	
Macro-Gap	SF13-04-15
Issue/Gap Description	Addressing the additional requirements in CSA N1600-14. There are a number of detailed additional requirements in CSA N1600-14 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: .... • processes for deviating from emergency response plans or recovery plans. .... Given that CSA N1600-14 is likely to be substantially revised in the short term, a phased approach should be taken to its detailed review for elements that need to be addressed by Bruce Power.

Gap #	SF13_CSA N1600-14_4.5.12_16
Document ID	CSA N1600-14
Article/Clause	4.5.12
Requirement Assessed	The emergency response plan includes a process for deviation from the plan and who can authorize this
Macro-Gap	SF13-02-16
Issue/Gap Description	CSA N1600 requires that the emergency response plan includes a process for deviation from the plan and who can authorize this. No such requirement is specified in CNSC REGDOC-2.10.1.

Gap #	SF13_CSA N1600-14_4.5.2_15
Document ID	CSA N1600-14
Article/Clause	4.5.2 Nuclear emergency response plan development
Requirement Assessed	
Macro-Gap	SF13-04-15
Issue/Gap Description	Addressing the additional requirements in CSA N1600-14. There are a number of detailed additional requirements in CSA N1600-14 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: • processes for deviating from emergency response plans or recovery plans. Given that CSA N1600-14 is likely to be substantially revised in the short term, a phased approach should be taken to its detailed review for elements that need to be addressed by Bruce Power.

Gap #	SF13_CSA N1600-14_4.5.2_16
Document ID	CSA N1600-14
Article/Clause	4.5.2
Requirement Assessed	The emergency response plan includes a process for deviation from the plan and who can authorize this
Macro-Gap	SF13-02-16
Issue/Gap Description	CSA N1600 requires that the emergency response plan includes a process for deviation from the plan and who can authorize this. No such requirement is specified in CNSC REGDOC-2.10.1.

Gap #	SF13_CSA N1600-14_5.4_15
Document ID	CSA N1600-14
Article/Clause	5.4 Deviation from the nuclear emergency response plan
Requirement Assessed	In the event that the organization needs to deviate from the nuclear emergency response plan (see Clause 4.5.12), the individual with authority shall authorize a) a deviation from the nuclear emergency response plan; b) the type of deviation required for the response efforts; and c) the resources (i.e., human, physical, informational, and financial) required to support the proposed deviation.
Macro-Gap	SF13-04-15
Issue/Gap Description	Addressing the additional requirements in CSA N1600-14. There are a number of detailed additional requirements in CSA N1600-14 that would need to be addressed for the implementation of the current version of the standard. The more significant of these include: .... • processes for deviating from emergency response plans or recovery plans. .... Given that CSA N1600-14 is likely to be substantially revised in the short term, a phased approach should be taken to its detailed review for elements that need to be addressed by Bruce Power.

Gap #	SF13_CSA N1600-14_5.4_16
Document ID	CSA N1600-14
Article/Clause	5.4
Requirement Assessed	A process for deviating from a recovery plan, and who can authorize this.
Macro-Gap	SF13-02-16
Issue/Gap Description	CSA N1600 requires a process for deviating from a recovery plan, and who can authorize this. CNSC REGDOC-2.10.1 does not contain such requirements.

Gap #	SF13_SF13 RT 2015_7.4_15
Document ID	SF13 RT 2015
Article/Clause	7.4
Requirement Assessed	Performance Indicators Performance indicators are defined as data that are sensitive to and/or signals changes in the performance of systems, components, or programs. In accordance with S-99 [65], Bruce Power reports on three Performance indicators (PIs) related to Emergency Preparedness for radiological emergencies: Radiological Emergencies Performance Index which provides an indication of the percentage of performance opportunities successfully demonstrated during drills, exercises or events for the past 8 quarters. Emergency Response Organization (ERO) Drill Participation Index - which provides an indication of the participation rate of key ERO personnel in drills, exercises or events calculated for an 8-quarter rolling average. Emergency Response Resources Completion Index which provides a measure of the completion rate of scheduled preventative maintenance.
Macro-Gap	SF13-01-15
Issue/Gap Description	ERO Drill participation rate-Except for the last 2 quarters reviewed, the ERO Drill: -Participation Index fluctuates between “satisfactory” and “improvement needed”, largely influenced by the corporate exercise schedule. ERO drill participation rate is thus identified as a gap.

Gap #	SF13_SF13 RT 2016_5.1_16
Document ID	SF13 RT 2016
Article/Clause	5.1
Requirement Assessed	An overall review will be performed to check that emergency planning at the plant continues to be satisfactory and to check that emergency plans (EPs) are maintained in accordance with current safety analyses, accident mitigation studies and good practices.
Macro-Gap	SF13-01-16
Issue/Gap Description	ERO staff selection-Audit findings from AU-2014-00005 (See Section 7.2) and issues identified in self-assessment SA-TRGD-2014-06 [35], represent a recurring problem with staff selection for the ERO organization

Gap #	SF13_SF13 RT 2016_5.1_16
Document ID	SF13 RT 2016
Article/Clause	5.1
Requirement Assessed	An overall review will be performed to check that emergency planning at the plant continues to be satisfactory and to check that emergency plans (EPs) are maintained in accordance with current safety analyses, accident mitigation studies and good practices.
Macro-Gap	SF13-01-16
Issue/Gap Description	As indicated in the Bruce Power transition plan for full compliance with REGDOC-2.10.1 [71], there is a need to complete the On-Site/Off-Site Emergency Response Communications Project to ensure that two independent means of communication are available to all emergency centres

Gap #	SF13_SF13 RT 2016_5.3.1_16
Document ID	SF13 RT 2016
Article/Clause	5.3.1
Requirement Assessed	Evaluate the adequacy of on-site equipment and facilities for emergencies.
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: -the number of electronic personal dosimeters dedicated to emergency response personnel

Gap #	SF13_SF13 RT 2016_5.3.1_16
Document ID	SF13 RT 2016
Article/Clause	5.3.1
Requirement Assessed	Evaluate the adequacy of on-site equipment and facilities for emergencies.
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: -potential delays in obtaining personal protective equipment from stores if access is impeded

Gap #	SF13_SF13 RT 2016_5.3.1_16
Document ID	SF13 RT 2016
Article/Clause	5.3.1
Requirement Assessed	Evaluate the adequacy of on-site equipment and facilities for emergencies.
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: -lack of severe accident dispersion calculations; and potential errors from the use of manual accounting method for centre of site staff during emergencies or site evacuation

Gap #	SF13_SF13 RT 2016_5.3.1_16
Document ID	SF13 RT 2016
Article/Clause	5.3.1
Requirement Assessed	Evaluate the adequacy of on-site equipment and facilities for emergencies.
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: -confirming worker safety should parts of the plant become uninhabitable flowing a four unit severe accident

Gap #	SF13_SF13 RT 2016_5.3.2_16
Document ID	SF13 RT 2016
Article/Clause	5.3.2
Requirement Assessed	Evaluate the adequacy of on-site technical and operational support centres
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: -radiation protection for EMC staff may not be sufficient. There is lack of a filtered ventilation system, and although the EMC can be relocated, this may delay the emergency response.

Gap #	SF13_SF13 RT 2016_5.3.3_16
Document ID	SF13 RT 2016
Article/Clause	5.3.3
Requirement Assessed	Evaluate the efficiency of communications in the event of an emergency, in particular the interaction with organizations outside the plant.
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: -there is lack of specific public address system announcements for multi-unit severe accidents.

Gap #	SF13_SF13 RT 2016_5.3.3_16
Document ID	SF13 RT 2016
Article/Clause	5.3.3
Requirement Assessed	Evaluate the efficiency of communications in the event of an emergency, in particular the interaction with organizations outside the plant.
Macro-Gap	SF13-03-16
Issue/Gap Description	As part of the OSART review the following issue was identified with respect to ERP, provisions and implementation: procedural guidance for shift managers to prioritize emergency classification could potentially lead to a delay in classifying an emergency and off-site notification.

Gap #	SF13_SF13 RT 2016_7.1_16
Document ID	SF13 RT 2016
Article/Clause	7.1
Requirement Assessed	Self-Assessments
Macro-Gap	SF13-01-16
Issue/Gap Description	Mutual Assist Response Team (MART) response timing-A timely MART response issue is evident from the drill and exercise results

Gap #	SF13_SF13 RT 2016_7.2.1.1_16
Document ID	SF13 RT 2016
Article/Clause	7.2.1.1
Requirement Assessed	AU-2014-00005, Nuclear Emergency Response Plan
Macro-Gap	SF13-01-16
Issue/Gap Description	Audit Follow-Up- Completion of outstanding corrective actions from 7.2.1.1. AU-2014-00005

Gap #	SF13_SF13 RT 2016_7.4_16
Document ID	SF13 RT 2016
Article/Clause	7.4
Requirement Assessed	Performance Indicators
Macro-Gap	SF13-01-16
Issue/Gap Description	ERO Drill participation rate-Except for the last 2 quarters reviewed, the ERO Drill: -Participation Index fluctuates between “satisfactory” and “improvement needed”, largely influenced by the corporate exercise schedule. ERO drill participation rate is thus identified as a gap.

Gap #	SF14_CSA N288.3.4-13_8.9_15
Document ID	CSA N288.3.4-13
Article/Clause	8.9
Requirement Assessed	Laboratory testing of adsorbent media
Macro-Gap	SF14-01-15
Issue/Gap Description	Requirement of pre service and in-service testing of adsorbent media (activated carbon) not met.

Gap #	SF14_CSA N288.3.4-13_10_15
Document ID	CSA N288.3.4-13
Article/Clause	10
Requirement Assessed	Quality assurance and quality control
Macro-Gap	SF14-01-15
Issue/Gap Description	Absence of appropriate QA/QC guidance in Bruce Power governing documents.

Gap #	SF14_CSA N288.3.4-13_11_15
Document ID	CSA N288.3.4-13
Article/Clause	11
Requirement Assessed	Reporting, review, and auditing
Macro-Gap	SF14-01-15
Issue/Gap Description	No record found of reviews (such as self-assessments) or independent audits of the air-cleaning system performance testing program.

Gap #	SF14_CSA N288.3.4-13_12_15
Document ID	CSA N288.3.4-13
Article/Clause	12
Requirement Assessed	Staff qualification and training
Macro-Gap	SF14-01-15
Issue/Gap Description	The following requirements of the Standard are not met: - the operator shall define the qualifications, and - if work is contracted out, documentation shall be available to demonstrate that the contract personnel have equivalent requisite qualifications.

Gap #	SF14_CSA N288.3.4-13_13_15
Document ID	CSA N288.3.4-13
Article/Clause	13
Requirement Assessed	Documentation
Macro-Gap	SF14-01-15
Issue/Gap Description	Much of the required testing program documentation is missing.

Gap #	SF14_CSA N288.3.4-13_8.5_15
Document ID	CSA N288.3.4-13
Article/Clause	8.5
Requirement Assessed	Air flow and pressure measurements
Macro-Gap	SF14-01-15
Issue/Gap Description	The absence of ongoing air pressure measurements.

Gap #	SF15_SF15 RT_5.2.2_15
Document ID	SF15 RT
Article/Clause	5.2.2
Requirement Assessed	2. RP Equipment and Instrumentation for Radiation Monitoring; 2.2 Radiation Protection Program Review
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. There are gaps in the effective implementation of the RP instrumentation program in order to maintain the fixed RP instrumentation (specifically FAGMs and whole-body contamination monitors) in good working condition.

Gap #	SF15_SF15 RT_5.2.2_15
Document ID	SF15 RT
Article/Clause	5.2.2
Requirement Assessed	2. RP Equipment and Instrumentation for Radiation Monitoring; 2.2 Radiation Protection Program Review
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. There is no documented lifecycle management process for the FAGMs system.

Gap #	SF15_SF15 RT_5.4.1_15
Document ID	SF15 RT
Article/Clause	5.4.1
Requirement Assessed	4. RP Organization and Administration; 4.2 Radiation Protection Program Review
Macro-Gap	SF15-04-15
Issue/Gap Description	Through the assessment of the Bruce Power RP Program and supporting procedures, it was observed that RP is not effectively using the Document Change Request (DCR) process to effectively implement changes to processes and/or procedures to improve the RP program as shown in self assessments and audits.

Gap #	SF15_WANO GL 2004-01-R1_VI.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C2.
Requirement Assessed	a. Automatic contamination monitors Automatic whole-body contamination monitors are, in most applications, superior to manual frisking. The whole-body monitors are particularly useful during surveying for discrete radioactive particles, because of the difficulty of detecting particles by hand frisking. Use whole-body contamination monitors at the exits from the primary RCA. Beta contamination monitors should detect levels at the average beta energy of the station radionuclide mix equivalent to 5,000 dpm at a distance from the detector equivalent to the location of the individual being monitored. Based on the configuration of the whole-body contamination monitor, the detector location, and the body part being monitored, this distance may be on contact with the grating above the foot detector and up to 3 inches from some other detectors. Establish alarm set points as low as practical, considering the presence of difficult-to-detect isotopes in the station radionuclide mix. At least daily, perform a response check on each detector, using a source with activity at the desired set point for the alarm and reasonably approximating the station isotopic mix. If a site chooses to response-check its RCA release instrumentation at a frequency other than "every detector, every day," the position should be well documented and include at least the following elements: o Instrument type and the location. Testing for both trains of detectors should be documented for instrumentation with dual detection capabilities (such as gas flow proportional detectors and gamma scintillation detectors). o Performance of deliberate failure tests to verify an instrument will remove itself from service prior to becoming ineffective o A formal process to evaluate and document instrumentation hardware or software modifications against initial testing, to ensure the monitor continues to function as expected o Testing to document as-found and as-left conditions to determine if the position should be reevaluated Periodically, whole-body contamination monitors should be challenged in the normal operating mode using a smear source representative of the station nuclide mix to determine their reliability and sensitivity. If the monitor does not have the ability to account for radon, have procedures in place to evaluate alarms for short-lived or natural radioactivity. Install gamma-sensitive portal monitors at the RCA exit to increase the likelihood of detecting contamination primarily composed of activated corrosion products that has proven difficult to detect with many types of automatic whole-body contamination monitors. Use portal monitors in a pause mode, and optimise their detection capability. Establish the alarm set point as low as practical, considering ambient background radiation, the negative consequences of false-positive alarms and reasonable

	egress times. Portal monitors located at the RCA exit should have alarm set points that correspond to 1111 1296 Bq (30-35 nanocuries) at Co-60 or about 2778-2963 Bq (75-80 nanocuries) of Cs-137. Gamma-sensitive portal monitors are not needed at the RCA exit if the whole-body contamination monitors incorporate a plastic scintillator or other detector capable of monitoring gamma radiation. At least each day an instrument is in use, perform a response check using a radioactive source with an activity appropriate for the alarm set point. Install gamma-sensitive portal monitors at the protected area exit as a final barrier, to increase the likelihood of detecting contamination that may have been inadvertently released from the RCA. Alarm set points should be based on station-specific isotopic mix and environment. If contamination monitor alarm set points are standardised across a multi-site fleet, use the set points from the station with the most limiting radionuclide mix, rather than an average among the sites. b. Hand frisking techniques The detectable quantity for a direct frisk with a thin-window Geiger-Mueller detector is nominally 100 counts per minute (cpm) above background with the background reading less than 300 cpm (a background reading of less than 200 cpm should be used if surveying for release of personnel). A minimally acceptable whole-body frisk requires two or three minutes. If background count rates below 300 cpm cannot be reasonably achieved at the desired monitoring location, frisk to check for gross contamination and perform a final frisk at a more remote location with acceptable background levels. Shielded frisking booths may be provided in high- background areas. Hand frisking should not be used for the release of personnel from the station without specific approval of the radiological protection manager, because of process difficulties and sensitivity. c. Contamination areas and radiologically controlled areas All personnel perform, as a minimum, a hand-and-foot frisk as soon as practical on exiting a contaminated area. When personnel exit a highly contaminated area (for example, greater than 100,000 dpm/100 cm2) or a discrete radioactive particle area, a whole-body frisk is done as soon as possible. In addition, a whole-body frisk using the whole- body contamination monitor or a frisker is performed before personnel put on any clothing not worn in the contaminated area. This ensures clothing does not lessen the sensitivity of the frisking process by shielding beta radiation. All persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole- body contamination monitor prior to exiting the RCA. It may not be practical to install whole-body contamination monitors and gamma-sensitive portal monitors at satellite RCAs, such as warehouses or radioactive material storage facilities. In these instances, personnel perform a survey using a whole-body contamination monitor or a hand-and- foot frisk upon leaving the satellite RCA and proceed to the nearest whole-body contamination monitor and gamma-sensitive portal monitor. Contamination monitoring requirements are clearly posted at the exit from satellite RCAs.

	Exiting any posted RCA or radioactive material storage area (RMA) requires personnel to monitor for contamination as specified above, with the following exceptions: o If the area is an RCA only because of dose rates and there are no radioactive material storage containers or contamination sources in the area, contamination monitoring is not required. o Personnel and material monitoring requirements for independent spent fuel storage installations (ISFSIs) should be the same as the primary RCA while fuel loading activities are in progress. After individual campaigns have been completed, ISFSI areas are exempt from monitoring because the contamination source is sealed within the certified container. o On a temporary basis, a satellite area posted as an RCA or RMA may be released from the need for contamination monitoring on each exit from the area. However, there must be no open contaminated areas or material, radiation protection coverage must be provided, and a sufficient survey of the area must be periodically performed (for example, each shift) to ensure no contamination is present. Provide equivalent contamination monitoring capability, as at the RCA exit, for personnel entering non-RCA clean areas inside the RCA where eating and drinking are allowed. For example, provide a whole-body contamination monitor and gamma tool monitor equivalent to those used at the RCA exit. d. Personnel Contamination Event A personnel contamination event is when an individual is contaminated greater than or equal to 100 counts per minute above background on the skin, clothing, or modesty garment, glasses, lanyard, shoes or hardhat. The highest contact frisker reading should be used to classify the personnel contamination event.
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. Technical Basis for whole-body monitor alarm test frequency is not available in formal documentation.

Gap #	SF15_WANO GL 2004-01-R1_VI.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C2.
Requirement Assessed	a. Automatic contamination monitors Automatic whole-body contamination monitors are, in most applications, superior to manual frisking. The whole-body monitors are particularly useful during surveying for discrete radioactive particles, because of the difficulty of detecting particles by hand frisking. Use whole-body contamination monitors at the exits from the primary RCA. Beta contamination monitors should detect levels at the average beta energy of the station radionuclide mix equivalent to 5,000 dpm at a distance from the detector equivalent to the location of the individual being monitored. Based on the configuration of the whole-body contamination monitor, the detector location, and the body part being monitored, this distance may be on contact with the grating above the foot detector and up to 3 inches from some other detectors. Establish alarm set points as low as practical, considering the presence of difficult-to-detect isotopes in the station radionuclide mix. At least daily, perform a response check on each detector, using a source with activity at the desired set point for the alarm and reasonably approximating the station isotopic mix. If a site chooses to response-check its RCA release instrumentation at a frequency other than "every detector, every day," the position should be well documented and include at least the following elements: o Instrument type and the location. Testing for both trains of detectors should be documented for instrumentation with dual detection capabilities (such as gas flow proportional detectors and gamma scintillation detectors). o Performance of deliberate failure tests to verify an instrument will remove itself from service prior to becoming ineffective o A formal process to evaluate and document instrumentation hardware or software modifications against initial testing, to ensure the monitor continues to function as expected o Testing to document as-found and as-left conditions to determine if the position should be reevaluated Periodically, whole-body contamination monitors should be challenged in the normal operating mode using a smear source representative of the station nuclide mix to determine their reliability and sensitivity. If the monitor does not have the ability to account for radon, have procedures in place to evaluate alarms for short-lived or natural radioactivity. Install gamma-sensitive portal monitors at the RCA exit to increase the likelihood of detecting contamination primarily composed of activated corrosion products that has proven difficult to detect with many types of automatic whole-body contamination monitors. Use portal monitors in a pause mode, and optimise their detection capability. Establish the alarm set point as low as practical, considering ambient background radiation, the negative consequences of false-positive alarms and reasonable

	egress times. Portal monitors located at the RCA exit should have alarm set points that correspond to 1111 1296 Bq (30-35 nanocuries) at Co-60 or about 2778-2963 Bq (75-80 nanocuries) of Cs-137. Gamma-sensitive portal monitors are not needed at the RCA exit if the whole-body contamination monitors incorporate a plastic scintillator or other detector capable of monitoring gamma radiation. At least each day an instrument is in use, perform a response check using a radioactive source with an activity appropriate for the alarm set point. Install gamma-sensitive portal monitors at the protected area exit as a final barrier, to increase the likelihood of detecting contamination that may have been inadvertently released from the RCA. Alarm set points should be based on station-specific isotopic mix and environment. If contamination monitor alarm set points are standardised across a multi-site fleet, use the set points from the station with the most limiting radionuclide mix, rather than an average among the sites. b. Hand frisking techniques The detectable quantity for a direct frisk with a thin-window Geiger-Mueller detector is nominally 100 counts per minute (cpm) above background with the background reading less than 300 cpm (a background reading of less than 200 cpm should be used if surveying for release of personnel). A minimally acceptable whole-body frisk requires two or three minutes. If background count rates below 300 cpm cannot be reasonably achieved at the desired monitoring location, frisk to check for gross contamination and perform a final frisk at a more remote location with acceptable background levels. Shielded frisking booths may be provided in high- background areas. Hand frisking should not be used for the release of personnel from the station without specific approval of the radiological protection manager, because of process difficulties and sensitivity. c. Contamination areas and radiologically controlled areas All personnel perform, as a minimum, a hand-and-foot frisk as soon as practical on exiting a contaminated area. When personnel exit a highly contaminated area (for example, greater than 100,000 dpm/100 cm2) or a discrete radioactive particle area, a whole-body frisk is done as soon as possible. In addition, a whole-body frisk using the whole- body contamination monitor or a frisker is performed before personnel put on any clothing not worn in the contaminated area. This ensures clothing does not lessen the sensitivity of the frisking process by shielding beta radiation. All persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole- body contamination monitor prior to exiting the RCA. It may not be practical to install whole-body contamination monitors and gamma-sensitive portal monitors at satellite RCAs, such as warehouses or radioactive material storage facilities. In these instances, personnel perform a survey using a whole-body contamination monitor or a hand-and- foot frisk upon leaving the satellite RCA and proceed to the nearest whole-body contamination monitor and gamma-sensitive portal monitor. Contamination monitoring requirements are clearly posted at the exit from satellite RCAs.

	Exiting any posted RCA or radioactive material storage area (RMA) requires personnel to monitor for contamination as specified above, with the following exceptions: o If the area is an RCA only because of dose rates and there are no radioactive material storage containers or contamination sources in the area, contamination monitoring is not required. o Personnel and material monitoring requirements for independent spent fuel storage installations (ISFSIs) should be the same as the primary RCA while fuel loading activities are in progress. After individual campaigns have been completed, ISFSI areas are exempt from monitoring because the contamination source is sealed within the certified container. o On a temporary basis, a satellite area posted as an RCA or RMA may be released from the need for contamination monitoring on each exit from the area. However, there must be no open contaminated areas or material, radiation protection coverage must be provided, and a sufficient survey of the area must be periodically performed (for example, each shift) to ensure no contamination is present. Provide equivalent contamination monitoring capability, as at the RCA exit, for personnel entering non-RCA clean areas inside the RCA where eating and drinking are allowed. For example, provide a whole-body contamination monitor and gamma tool monitor equivalent to those used at the RCA exit. d. Personnel Contamination Event A personnel contamination event is when an individual is contaminated greater than or equal to 100 counts per minute above background on the skin, clothing, or modesty garment, glasses, lanyard, shoes or hardhat. The highest contact frisker reading should be used to classify the personnel contamination event.
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. Technical Basis for use of a check source for whole-body monitors that approximates the station isotopic mix is not available in formal documentation.

Gap #	SF15_WANO GL 2004-01-R1_VI.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C2.
Requirement Assessed	a. Automatic contamination monitors Automatic whole-body contamination monitors are, in most applications, superior to manual frisking. The whole-body monitors are particularly useful during surveying for discrete radioactive particles, because of the difficulty of detecting particles by hand frisking. Use whole-body contamination monitors at the exits from the primary RCA. Beta contamination monitors should detect levels at the average beta energy of the station radionuclide mix equivalent to 5,000 dpm at a distance from the detector equivalent to the location of the individual being monitored. Based on the configuration of the whole-body contamination monitor, the detector location, and the body part being monitored, this distance may be on contact with the grating above the foot detector and up to 3 inches from some other detectors. Establish alarm set points as low as practical, considering the presence of difficult-to-detect isotopes in the station radionuclide mix. At least daily, perform a response check on each detector, using a source with activity at the desired set point for the alarm and reasonably approximating the station isotopic mix. If a site chooses to response-check its RCA release instrumentation at a frequency other than "every detector, every day," the position should be well documented and include at least the following elements: o Instrument type and the location. Testing for both trains of detectors should be documented for instrumentation with dual detection capabilities (such as gas flow proportional detectors and gamma scintillation detectors). o Performance of deliberate failure tests to verify an instrument will remove itself from service prior to becoming ineffective o A formal process to evaluate and document instrumentation hardware or software modifications against initial testing, to ensure the monitor continues to function as expected o Testing to document as-found and as-left conditions to determine if the position should be reevaluated Periodically, whole-body contamination monitors should be challenged in the normal operating mode using a smear source representative of the station nuclide mix to determine their reliability and sensitivity. If the monitor does not have the ability to account for radon, have procedures in place to evaluate alarms for short-lived or natural radioactivity. Install gamma-sensitive portal monitors at the RCA exit to increase the likelihood of detecting contamination primarily composed of activated corrosion products that has proven difficult to detect with many types of automatic whole-body contamination monitors. Use portal monitors in a pause mode, and optimise their detection capability. Establish the alarm set point as low as practical, considering ambient background radiation, the negative consequences of false-positive alarms and reasonable

	egress times. Portal monitors located at the RCA exit should have alarm set points that correspond to 1111 1296 Bq (30-35 nanocuries) at Co-60 or about 2778-2963 Bq (75-80 nanocuries) of Cs-137. Gamma-sensitive portal monitors are not needed at the RCA exit if the whole-body contamination monitors incorporate a plastic scintillator or other detector capable of monitoring gamma radiation. At least each day an instrument is in use, perform a response check using a radioactive source with an activity appropriate for the alarm set point. Install gamma-sensitive portal monitors at the protected area exit as a final barrier, to increase the likelihood of detecting contamination that may have been inadvertently released from the RCA. Alarm set points should be based on station-specific isotopic mix and environment. If contamination monitor alarm set points are standardised across a multi-site fleet, use the set points from the station with the most limiting radionuclide mix, rather than an average among the sites. b. Hand frisking techniques The detectable quantity for a direct frisk with a thin-window Geiger-Mueller detector is nominally 100 counts per minute (cpm) above background with the background reading less than 300 cpm (a background reading of less than 200 cpm should be used if surveying for release of personnel). A minimally acceptable whole-body frisk requires two or three minutes. If background count rates below 300 cpm cannot be reasonably achieved at the desired monitoring location, frisk to check for gross contamination and perform a final frisk at a more remote location with acceptable background levels. Shielded frisking booths may be provided in high- background areas. Hand frisking should not be used for the release of personnel from the station without specific approval of the radiological protection manager, because of process difficulties and sensitivity. c. Contamination areas and radiologically controlled areas All personnel perform, as a minimum, a hand-and-foot frisk as soon as practical on exiting a contaminated area. When personnel exit a highly contaminated area (for example, greater than 100,000 dpm/100 cm2) or a discrete radioactive particle area, a whole-body frisk is done as soon as possible. In addition, a whole-body frisk using the whole- body contamination monitor or a frisker is performed before personnel put on any clothing not worn in the contaminated area. This ensures clothing does not lessen the sensitivity of the frisking process by shielding beta radiation. All persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole- body contamination monitor prior to exiting the RCA. It may not be practical to install whole-body contamination monitors and gamma-sensitive portal monitors at satellite RCAs, such as warehouses or radioactive material storage facilities. In these instances, personnel perform a survey using a whole-body contamination monitor or a hand-and- foot frisk upon leaving the satellite RCA and proceed to the nearest whole-body contamination monitor and gamma-sensitive portal monitor. Contamination monitoring requirements are clearly posted at the exit from satellite RCAs.

	Exiting any posted RCA or radioactive material storage area (RMA) requires personnel to monitor for contamination as specified above, with the following exceptions: o If the area is an RCA only because of dose rates and there are no radioactive material storage containers or contamination sources in the area, contamination monitoring is not required. o Personnel and material monitoring requirements for independent spent fuel storage installations (ISFSIs) should be the same as the primary RCA while fuel loading activities are in progress. After individual campaigns have been completed, ISFSI areas are exempt from monitoring because the contamination source is sealed within the certified container. o On a temporary basis, a satellite area posted as an RCA or RMA may be released from the need for contamination monitoring on each exit from the area. However, there must be no open contaminated areas or material, radiation protection coverage must be provided, and a sufficient survey of the area must be periodically performed (for example, each shift) to ensure no contamination is present. Provide equivalent contamination monitoring capability, as at the RCA exit, for personnel entering non-RCA clean areas inside the RCA where eating and drinking are allowed. For example, provide a whole-body contamination monitor and gamma tool monitor equivalent to those used at the RCA exit. d. Personnel Contamination Event A personnel contamination event is when an individual is contaminated greater than or equal to 100 counts per minute above background on the skin, clothing, or modesty garment, glasses, lanyard, shoes or hardhat. The highest contact frisker reading should be used to classify the personnel contamination event.
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. There is no programmatic requirement to perform routine tests to challenge whole-body contamination monitors using a smear source representative of the station nuclide mix to determine the reliability and sensitivity.

Gap #	SF15_WANO GL 2004-01-R1_VI.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C2.
Requirement Assessed	a. Automatic contamination monitors Automatic whole-body contamination monitors are, in most applications, superior to manual frisking. The whole-body monitors are particularly useful during surveying for discrete radioactive particles, because of the difficulty of detecting particles by hand frisking. Use whole-body contamination monitors at the exits from the primary RCA. Beta contamination monitors should detect levels at the average beta energy of the station radionuclide mix equivalent to 5,000 dpm at a distance from the detector equivalent to the location of the individual being monitored. Based on the configuration of the whole-body contamination monitor, the detector location, and the body part being monitored, this distance may be on contact with the grating above the foot detector and up to 3 inches from some other detectors. Establish alarm set points as low as practical, considering the presence of difficult-to-detect isotopes in the station radionuclide mix. At least daily, perform a response check on each detector, using a source with activity at the desired set point for the alarm and reasonably approximating the station isotopic mix. If a site chooses to response-check its RCA release instrumentation at a frequency other than "every detector, every day," the position should be well documented and include at least the following elements: o Instrument type and the location. Testing for both trains of detectors should be documented for instrumentation with dual detection capabilities (such as gas flow proportional detectors and gamma scintillation detectors). o Performance of deliberate failure tests to verify an instrument will remove itself from service prior to becoming ineffective o A formal process to evaluate and document instrumentation hardware or software modifications against initial testing, to ensure the monitor continues to function as expected o Testing to document as-found and as-left conditions to determine if the position should be reevaluated Periodically, whole-body contamination monitors should be challenged in the normal operating mode using a smear source representative of the station nuclide mix to determine their reliability and sensitivity. If the monitor does not have the ability to account for radon, have procedures in place to evaluate alarms for short-lived or natural radioactivity. Install gamma-sensitive portal monitors at the RCA exit to increase the likelihood of detecting contamination primarily composed of activated corrosion products that has proven difficult to detect with many types of automatic whole-body contamination monitors. Use portal monitors in a pause mode, and optimise their detection capability. Establish the alarm set point as low as practical, considering ambient background radiation, the negative consequences of false-positive alarms and reasonable

	egress times. Portal monitors located at the RCA exit should have alarm set points that correspond to 1111 1296 Bq (30-35 nanocuries) at Co-60 or about 2778-2963 Bq (75-80 nanocuries) of Cs-137. Gamma-sensitive portal monitors are not needed at the RCA exit if the whole-body contamination monitors incorporate a plastic scintillator or other detector capable of monitoring gamma radiation. At least each day an instrument is in use, perform a response check using a radioactive source with an activity appropriate for the alarm set point. Install gamma-sensitive portal monitors at the protected area exit as a final barrier, to increase the likelihood of detecting contamination that may have been inadvertently released from the RCA. Alarm set points should be based on station-specific isotopic mix and environment. If contamination monitor alarm set points are standardised across a multi-site fleet, use the set points from the station with the most limiting radionuclide mix, rather than an average among the sites. b. Hand frisking techniques The detectable quantity for a direct frisk with a thin-window Geiger-Mueller detector is nominally 100 counts per minute (cpm) above background with the background reading less than 300 cpm (a background reading of less than 200 cpm should be used if surveying for release of personnel). A minimally acceptable whole-body frisk requires two or three minutes. If background count rates below 300 cpm cannot be reasonably achieved at the desired monitoring location, frisk to check for gross contamination and perform a final frisk at a more remote location with acceptable background levels. Shielded frisking booths may be provided in high- background areas. Hand frisking should not be used for the release of personnel from the station without specific approval of the radiological protection manager, because of process difficulties and sensitivity. c. Contamination areas and radiologically controlled areas All personnel perform, as a minimum, a hand-and-foot frisk as soon as practical on exiting a contaminated area. When personnel exit a highly contaminated area (for example, greater than 100,000 dpm/100 cm2) or a discrete radioactive particle area, a whole-body frisk is done as soon as possible. In addition, a whole-body frisk using the whole- body contamination monitor or a frisker is performed before personnel put on any clothing not worn in the contaminated area. This ensures clothing does not lessen the sensitivity of the frisking process by shielding beta radiation. All persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole- body contamination monitor prior to exiting the RCA. It may not be practical to install whole-body contamination monitors and gamma-sensitive portal monitors at satellite RCAs, such as warehouses or radioactive material storage facilities. In these instances, personnel perform a survey using a whole-body contamination monitor or a hand-and- foot frisk upon leaving the satellite RCA and proceed to the nearest whole-body contamination monitor and gamma-sensitive portal monitor. Contamination monitoring requirements are clearly posted at the exit from satellite RCAs.

	Exiting any posted RCA or radioactive material storage area (RMA) requires personnel to monitor for contamination as specified above, with the following exceptions: o If the area is an RCA only because of dose rates and there are no radioactive material storage containers or contamination sources in the area, contamination monitoring is not required. o Personnel and material monitoring requirements for independent spent fuel storage installations (ISFSIs) should be the same as the primary RCA while fuel loading activities are in progress. After individual campaigns have been completed, ISFSI areas are exempt from monitoring because the contamination source is sealed within the certified container. o On a temporary basis, a satellite area posted as an RCA or RMA may be released from the need for contamination monitoring on each exit from the area. However, there must be no open contaminated areas or material, radiation protection coverage must be provided, and a sufficient survey of the area must be periodically performed (for example, each shift) to ensure no contamination is present. Provide equivalent contamination monitoring capability, as at the RCA exit, for personnel entering non-RCA clean areas inside the RCA where eating and drinking are allowed. For example, provide a whole-body contamination monitor and gamma tool monitor equivalent to those used at the RCA exit. d. Personnel Contamination Event A personnel contamination event is when an individual is contaminated greater than or equal to 100 counts per minute above background on the skin, clothing, or modesty garment, glasses, lanyard, shoes or hardhat. The highest contact frisker reading should be used to classify the personnel contamination event.
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. There is no programmatic requirement for placement of gamma-detection capability (such as plastic scintillation detectors) at all radiologically-controlled area exits is not available in formal documentation.

Gap #	SF15_WANO GL 2004-01-R1_VI.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C2.
Requirement Assessed	a. Automatic contamination monitors Automatic whole-body contamination monitors are, in most applications, superior to manual frisking. The whole-body monitors are particularly useful during surveying for discrete radioactive particles, because of the difficulty of detecting particles by hand frisking. Use whole-body contamination monitors at the exits from the primary RCA. Beta contamination monitors should detect levels at the average beta energy of the station radionuclide mix equivalent to 5,000 dpm at a distance from the detector equivalent to the location of the individual being monitored. Based on the configuration of the whole-body contamination monitor, the detector location, and the body part being monitored, this distance may be on contact with the grating above the foot detector and up to 3 inches from some other detectors. Establish alarm set points as low as practical, considering the presence of difficult-to-detect isotopes in the station radionuclide mix. At least daily, perform a response check on each detector, using a source with activity at the desired set point for the alarm and reasonably approximating the station isotopic mix. If a site chooses to response-check its RCA release instrumentation at a frequency other than "every detector, every day," the position should be well documented and include at least the following elements: o Instrument type and the location. Testing for both trains of detectors should be documented for instrumentation with dual detection capabilities (such as gas flow proportional detectors and gamma scintillation detectors). o Performance of deliberate failure tests to verify an instrument will remove itself from service prior to becoming ineffective o A formal process to evaluate and document instrumentation hardware or software modifications against initial testing, to ensure the monitor continues to function as expected o Testing to document as-found and as-left conditions to determine if the position should be reevaluated Periodically, whole-body contamination monitors should be challenged in the normal operating mode using a smear source representative of the station nuclide mix to determine their reliability and sensitivity. If the monitor does not have the ability to account for radon, have procedures in place to evaluate alarms for short-lived or natural radioactivity. Install gamma-sensitive portal monitors at the RCA exit to increase the likelihood of detecting contamination primarily composed of activated corrosion products that has proven difficult to detect with many types of automatic whole-body contamination monitors. Use portal monitors in a pause mode, and optimise their detection capability. Establish the alarm set point as low as practical, considering ambient background radiation, the negative consequences of false-positive alarms and reasonable

	egress times. Portal monitors located at the RCA exit should have alarm set points that correspond to 1111 1296 Bq (30-35 nanocuries) at Co-60 or about 2778-2963 Bq (75-80 nanocuries) of Cs-137. Gamma-sensitive portal monitors are not needed at the RCA exit if the whole-body contamination monitors incorporate a plastic scintillator or other detector capable of monitoring gamma radiation. At least each day an instrument is in use, perform a response check using a radioactive source with an activity appropriate for the alarm set point. Install gamma-sensitive portal monitors at the protected area exit as a final barrier, to increase the likelihood of detecting contamination that may have been inadvertently released from the RCA. Alarm set points should be based on station-specific isotopic mix and environment. If contamination monitor alarm set points are standardised across a multi-site fleet, use the set points from the station with the most limiting radionuclide mix, rather than an average among the sites. b. Hand frisking techniques The detectable quantity for a direct frisk with a thin-window Geiger-Mueller detector is nominally 100 counts per minute (cpm) above background with the background reading less than 300 cpm (a background reading of less than 200 cpm should be used if surveying for release of personnel). A minimally acceptable whole-body frisk requires two or three minutes. If background count rates below 300 cpm cannot be reasonably achieved at the desired monitoring location, frisk to check for gross contamination and perform a final frisk at a more remote location with acceptable background levels. Shielded frisking booths may be provided in high- background areas. Hand frisking should not be used for the release of personnel from the station without specific approval of the radiological protection manager, because of process difficulties and sensitivity. c. Contamination areas and radiologically controlled areas All personnel perform, as a minimum, a hand-and-foot frisk as soon as practical on exiting a contaminated area. When personnel exit a highly contaminated area (for example, greater than 100,000 dpm/100 cm2) or a discrete radioactive particle area, a whole-body frisk is done as soon as possible. In addition, a whole-body frisk using the whole- body contamination monitor or a frisker is performed before personnel put on any clothing not worn in the contaminated area. This ensures clothing does not lessen the sensitivity of the frisking process by shielding beta radiation. All persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole- body contamination monitor prior to exiting the RCA. It may not be practical to install whole-body contamination monitors and gamma-sensitive portal monitors at satellite RCAs, such as warehouses or radioactive material storage facilities. In these instances, personnel perform a survey using a whole-body contamination monitor or a hand-and- foot frisk upon leaving the satellite RCA and proceed to the nearest whole-body contamination monitor and gamma-sensitive portal monitor. Contamination monitoring requirements are clearly posted at the exit from satellite RCAs.

	Exiting any posted RCA or radioactive material storage area (RMA) requires personnel to monitor for contamination as specified above, with the following exceptions: o If the area is an RCA only because of dose rates and there are no radioactive material storage containers or contamination sources in the area, contamination monitoring is not required. o Personnel and material monitoring requirements for independent spent fuel storage installations (ISFSIs) should be the same as the primary RCA while fuel loading activities are in progress. After individual campaigns have been completed, ISFSI areas are exempt from monitoring because the contamination source is sealed within the certified container. o On a temporary basis, a satellite area posted as an RCA or RMA may be released from the need for contamination monitoring on each exit from the area. However, there must be no open contaminated areas or material, radiation protection coverage must be provided, and a sufficient survey of the area must be periodically performed (for example, each shift) to ensure no contamination is present. Provide equivalent contamination monitoring capability, as at the RCA exit, for personnel entering non-RCA clean areas inside the RCA where eating and drinking are allowed. For example, provide a whole-body contamination monitor and gamma tool monitor equivalent to those used at the RCA exit. d. Personnel Contamination Event A personnel contamination event is when an individual is contaminated greater than or equal to 100 counts per minute above background on the skin, clothing, or modesty garment, glasses, lanyard, shoes or hardhat. The highest contact frisker reading should be used to classify the personnel contamination event.
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. Technical Basis for whole-body monitor alarm set points is not available in formal documentation.

Gap #	SF15_WANO GL 2004-01-R1_VI.C2._15
Document ID	WANO GL 2004-01-R1
Article/Clause	VI.C2.
Requirement Assessed	a. Automatic contamination monitors Automatic whole-body contamination monitors are, in most applications, superior to manual frisking. The whole-body monitors are particularly useful during surveying for discrete radioactive particles, because of the difficulty of detecting particles by hand frisking. Use whole-body contamination monitors at the exits from the primary RCA. Beta contamination monitors should detect levels at the average beta energy of the station radionuclide mix equivalent to 5,000 dpm at a distance from the detector equivalent to the location of the individual being monitored. Based on the configuration of the whole-body contamination monitor, the detector location, and the body part being monitored, this distance may be on contact with the grating above the foot detector and up to 3 inches from some other detectors. Establish alarm set points as low as practical, considering the presence of difficult-to-detect isotopes in the station radionuclide mix. At least daily, perform a response check on each detector, using a source with activity at the desired set point for the alarm and reasonably approximating the station isotopic mix. If a site chooses to response-check its RCA release instrumentation at a frequency other than "every detector, every day," the position should be well documented and include at least the following elements: o Instrument type and the location. Testing for both trains of detectors should be documented for instrumentation with dual detection capabilities (such as gas flow proportional detectors and gamma scintillation detectors). o Performance of deliberate failure tests to verify an instrument will remove itself from service prior to becoming ineffective o A formal process to evaluate and document instrumentation hardware or software modifications against initial testing, to ensure the monitor continues to function as expected o Testing to document as-found and as-left conditions to determine if the position should be reevaluated Periodically, whole-body contamination monitors should be challenged in the normal operating mode using a smear source representative of the station nuclide mix to determine their reliability and sensitivity. If the monitor does not have the ability to account for radon, have procedures in place to evaluate alarms for short-lived or natural radioactivity. Install gamma-sensitive portal monitors at the RCA exit to increase the likelihood of detecting contamination primarily composed of activated corrosion products that has proven difficult to detect with many types of automatic whole-body contamination monitors. Use portal monitors in a pause mode, and optimise their detection capability. Establish the alarm set point as low as practical, considering ambient background radiation, the negative consequences of false-positive alarms and reasonable

	egress times. Portal monitors located at the RCA exit should have alarm set points that correspond to 1111 1296 Bq (30-35 nanocuries) at Co-60 or about 2778-2963 Bq (75-80 nanocuries) of Cs-137. Gamma-sensitive portal monitors are not needed at the RCA exit if the whole-body contamination monitors incorporate a plastic scintillator or other detector capable of monitoring gamma radiation. At least each day an instrument is in use, perform a response check using a radioactive source with an activity appropriate for the alarm set point. Install gamma-sensitive portal monitors at the protected area exit as a final barrier, to increase the likelihood of detecting contamination that may have been inadvertently released from the RCA. Alarm set points should be based on station-specific isotopic mix and environment. If contamination monitor alarm set points are standardised across a multi-site fleet, use the set points from the station with the most limiting radionuclide mix, rather than an average among the sites. b. Hand frisking techniques The detectable quantity for a direct frisk with a thin-window Geiger-Mueller detector is nominally 100 counts per minute (cpm) above background with the background reading less than 300 cpm (a background reading of less than 200 cpm should be used if surveying for release of personnel). A minimally acceptable whole-body frisk requires two or three minutes. If background count rates below 300 cpm cannot be reasonably achieved at the desired monitoring location, frisk to check for gross contamination and perform a final frisk at a more remote location with acceptable background levels. Shielded frisking booths may be provided in high- background areas. Hand frisking should not be used for the release of personnel from the station without specific approval of the radiological protection manager, because of process difficulties and sensitivity. c. Contamination areas and radiologically controlled areas All personnel perform, as a minimum, a hand-and-foot frisk as soon as practical on exiting a contaminated area. When personnel exit a highly contaminated area (for example, greater than 100,000 dpm/100 cm2) or a discrete radioactive particle area, a whole-body frisk is done as soon as possible. In addition, a whole-body frisk using the whole- body contamination monitor or a frisker is performed before personnel put on any clothing not worn in the contaminated area. This ensures clothing does not lessen the sensitivity of the frisking process by shielding beta radiation. All persons, regardless of whether they entered a contaminated area, should monitor themselves for contamination with a whole- body contamination monitor prior to exiting the RCA. It may not be practical to install whole-body contamination monitors and gamma-sensitive portal monitors at satellite RCAs, such as warehouses or radioactive material storage facilities. In these instances, personnel perform a survey using a whole-body contamination monitor or a hand-and- foot frisk upon leaving the satellite RCA and proceed to the nearest whole-body contamination monitor and gamma-sensitive portal monitor. Contamination monitoring requirements are clearly posted at the exit from satellite RCAs.

	Exiting any posted RCA or radioactive material storage area (RMA) requires personnel to monitor for contamination as specified above, with the following exceptions: o If the area is an RCA only because of dose rates and there are no radioactive material storage containers or contamination sources in the area, contamination monitoring is not required. o Personnel and material monitoring requirements for independent spent fuel storage installations (ISFSIs) should be the same as the primary RCA while fuel loading activities are in progress. After individual campaigns have been completed, ISFSI areas are exempt from monitoring because the contamination source is sealed within the certified container. o On a temporary basis, a satellite area posted as an RCA or RMA may be released from the need for contamination monitoring on each exit from the area. However, there must be no open contaminated areas or material, radiation protection coverage must be provided, and a sufficient survey of the area must be periodically performed (for example, each shift) to ensure no contamination is present. Provide equivalent contamination monitoring capability, as at the RCA exit, for personnel entering non-RCA clean areas inside the RCA where eating and drinking are allowed. For example, provide a whole-body contamination monitor and gamma tool monitor equivalent to those used at the RCA exit. d. Personnel Contamination Event A personnel contamination event is when an individual is contaminated greater than or equal to 100 counts per minute above background on the skin, clothing, or modesty garment, glasses, lanyard, shoes or hardhat. The highest contact frisker reading should be used to classify the personnel contamination event.
Macro-Gap	SF15-03-15
Issue/Gap Description	The Technical Basis for RP instrumentation setpoints, locations and function checks is not provided in formal documentation. Technical Basis for portal monitor alarm set points is not available in formal documentation.

Appendix H – List of CARDs

CARD #	CA-0067
CARD Title	Fuel Channel Life Management
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-13380 / NK29-CORR-00531-13927 - ACTION ITEM 1407-4775: ANNUAL UPDATE ON APPROACH TO FITNESS-FOR-SERVICE ASSESSMENT FOR PRESSURE TUBES - UPDATE 5. Tracked under AI 1407-4775.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTNSAS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Results of Fuel Channel Life Management Projects will have an immediate impact on maintaining the design basis for fuel channels and reducing associated uncertainties associated with material properties
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining design basis, Column 2- Augments recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-025
GIO Title	Perform R&D in support of fuel channel life cycle management initiatives
CARD(s) Associated with this GIO	CA-0067
Gap(s) Associated with this CARD	SF04-IIP-003-14
Additional Information	The Fuel Channel Condition Assessment (FCCA) has been updated recently to include inspection information, understanding on pressure tube and spacer degradation mechanisms, and new development on the R&D program, received up to June 2015. The Fuel Channel Life Cycle Management Plan (FCLCMP) has been updated, and is now in the document B-LCM-31100-00001 (R000), Fuel Channel Life Cycle Management Plan. B-LCM-31100-00001 (R000) supercedes B-PLAN-31100-00001 (R005), Fuel Channel Life Cycle Management Plan. In addition to FCLCMP and FCCA, there are separate processes established with the CNSC to discuss key methodologies for pressure tube fitness-for-service evaluations and, ultimately, gain

	regulatory acceptance for use on Bruce Units. All these documents and processes are in place to establish the technical support and confidence in the continued operation of Bruce Units to their target life. The following processes to address these technical issues items are summarized as follows: 1) Probabilistic Core Assessment (PCA) - Bruce Power has been performing PCA for over 10 years and has establish a process to update the PCA on a 3-year cycle. The CNSC has expressed concerns about the "flaw removal" treatment in PCA and Bruce Power has provided a plan to address this issue. The CNSC also provided further technical issues related to PLBB and PCA to OPG and Bruce Power recently at a meeting on May 29, 2017 (Reference 2). These issues will be addressed within the existing process for PCA. 2) Probabilistic Leak-Before-Break (PLBB) Assessments - Bruce Power has adopted the PLBB methodology according to Clause 7.4.3.3 in CSA N285.8-15, which allows an integrated probabilistic core evaluation of crack initiation and LBB. The way PCA in item 1 is currently performed would meet the majority of the evaluation requirements. Additional calculations on the conditional probability of Break-Before-Leak will be included in further PCA update to meet all the requirements as per Clause 7.4.3.3. The CNSC provided technical issues related to PLBB and PCA to OPG and Bruce Power recently at a meeting on May 29, 2017 (CNSC e-Docs #5243387-v3 PPTX - Topics Related to PLBB and PCA). These issues will be addressed within the existing process for PLBB. 3) Probabilistic Fracture Protection (PFP) - There is a separate process in place with the CNSC to establish a regulatory position on the PFP methodology by end of 2017 as documented in NK21-CORR-00531-12921/NK29-CORR-00531-13384- Proposed Information Exchange on the Methodology of Probabilistic Fracture Protection Evaluation. The schedule is subject to change due to resource availability and additional work due to findings. 4) Probabilistic Pressure Tube to Calandria Tube Contact Assessments - The CNSC has previously accepted PT/CT contact disposition based on probabilistic methodology. A process was established with the CNSC to address the third-party review comments on the probabilistic methodology. There is also a CSA task group to develop the statistical blister cracking model for use in the probabilistic PT/CT contact assessment. 5) A model to assess Hydrided Region Overload for detected flaws - a short-term and a long-term plans to address the issue have been submitted to the CNSC in NK21-CORR-00531-13019/NK29-CORR-00531-13486 - Evaluation of Crack Initiation due to Hydrided Region Overload in Pressure Tube Flaws and NK21-CORR-00531-13414/NK29-CORR-00531-13961 - Evaluation of Crack Initiation due to Hydrided Region Overload in Pressure Tube Flaws. An industry workshop is planned for September 2017. 6) Pressure Tube Fracture Toughness - this is an ongoing issue being

	addressed by a CNSC Action Item 1407-4775 for which an annual update on the progress of the R&D program to address the CNSC and third-party review comments, is provided. The latest updated was provided in NK21-CORR-00531-13380/NK29-CORR-00531-13927 (Action Item 1407-4775: Annual Update on Approach to Fitness-for-Service Assessment for Pressure Tubes - Update 5). 7) Degradation of tight-fitting spacers - OPG and Bruce Power has an annual update meeting with the CNSC to provide updates on improved understanding of Inconel X-750 spacer degradation mechanisms and latest testing results. The next planned meeting is in September 2017. Recent testing on spacers removed from channel B8J18 has shown significant margin to the design load. An assessment is being performed to project spacer load carrying capacity to target life for applicable Bruce Units. An update on the status of these separate documents and processes will be included as part of the IIP periodic updates for monitoring the progress and implementation of the IIP.
References	NK21-CORR-00531-13680 / NK29-CORR-00531-14326 CNSC e-Docs #5243387-v3 NK21-CORR-00531-10978 / NK29-CORR-00531-11366 NK21-CORR-00531-11472 NK21-CORR-00531-12248 / NK29-CORR-00531-12672 NK21-CORR-00531-12618 / NK29-CORR-00531-13046 NK21-CORR-00531-12662 / NK29-CORR-00531-13098 NK21-CORR-00531-12921 / NK29-CORR-00531-13384 NK21-CORR-00531-13019 / NK29-CORR-00531-13486 NK21-CORR-00531-13380 / NK29-CORR-00531-13927 NK21-CORR-00531-13414 / NK29-CORR-00531-13961 NK29-CORR-00531-11868 AI 1407-4775

CARD #	CA-0071
CARD Title	SIP-30: BA U1/U2 Post RTS - Standby Generator Controls Replacement
CARD Description	Standby generator controls upgrade for SG1, SG2, SG3 & SG4 per AI 1207-3283 & Project 36527 - BA U1/U2 Post RTS - Standby Generator Controls Replacement.
Applicable Units	Bruce A
Alert Group	DPTPROJC
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Standby Generator Controls Replacement will have an immediate impact on maintaining the design basis of Standby Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Standby Generators. Column 2-Augments/recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Underway
Target Completion Date	17-Dec-21
GIO #	GIO-028
GIO Title	Upgrade Emergency and Standby Power Supplies
CARD(s) Associated with this GIO	CA-0071, CA-0073
Gap(s) Associated with this CARD	SF01-IIP-001-14
Additional Information	The Bruce A project scope has been expanded to include the final two Standby Generators (SG1 and SG2), which has been funded by the Units 3 and 4 Long Term Asset Management Plan. The milestone status of the project at Bruce A is as follows: • Detailed design of the equipment was completed and accepted in Q3 2016. • Panel manufacture will commence in Q4 2016. • Final Design Packages for the first Standby Generator, SG3, will be issued by Q1 2017. • Factory Acceptance Testing will be completed in Q3 2017 based on manufacturer-supplied schedules. • Based on the Work Management Long Range Cycle Plan, the four Bruce A Standby Generators have controls upgrades scheduled to start in the following quarters: SG3—Q4 2017

	SG4—Q3 2018 SG1 —Q32019 SG2—Q3 2020. The controls upgrades are being coordinated to occur at the same time as major SG turbine/generator equipment overhauls to allow for comprehensive system tuning during the commissioning and return to service. The Bruce B Standby Generator control upgrade project has completed SG7 and SG8. SG5 and SG6 continue to meet their reliability targets, with sufficient spare parts available to ensure future reliability. The next update on the progress of the Standby Generator control upgrade project will be provided by the end of 2017. This project is in the PMC Preparation Phase.
References	NK21-CORR-00531-13161 / NK29-CORR-00531-13649 NK21-CORR-00531-12449 / NK29-CORR-00531-12861 NK21-CORR-00531-11366 AI 1207-3283

CARD #	CA-0073
CARD Title	SIP-35: Emergency Power Generators 1 and 2 Upgrades
CARD Description	Emergency power generator controls upgrades for EPG1 (EPG2 complete) per AI 111402. This initiative is implemented under Project # 32003 - BB Emergency Power Generator (EPG) Controls Upgrade.
Applicable Units	Bruce B
Alert Group	DPTPROJC
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Emergency Power Generators 1 and 2 Upgrades will have an immediate impact on maintaining the design basis of Standby Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Emergency Power Generators 1 and 2. Column 2-Augments/recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Underway
Target Completion Date	22-Dec-17
GIO #	GIO-028
GIO Title	Upgrade Emergency and Standby Power Supplies
CARD(s) Associated with this GIO	CA-0071, CA-0073
Gap(s) Associated with this CARD	SF04-IIP-002-14
Additional Information	This project is in the PMC Execution Phase. SIP action is in progress. AI 111402 was closed based on the improved system health discussed in NK29-CORR-00531-12003, CNSC acceptance of the closure criteria is documented in NK29-CORR-00531-12077. Bruce Power is executing the Bruce B EPG System Health Improvement Plan, which includes upgrades to EPG1 and EPG2. The upgrade project is also expected to provide improvements in the reliability of EPG1 and EPG2, equivalent to that assumed in the Probabilistic Safety Analysis for EPG3. Bruce Power completed the EPG2 controls upgrade in the summer of 2015. The mechanical overhaul of EPG2 is currently in progress, and is expected to be completed in the fall of 2016. The EPG1 controls upgrade and mechanical

	overhaul will be executed in parallel and is presently scheduled to be completed in the summer of 2017. Bruce Power will continue to progress the EPG upgrade project, and will notify CNSC staff upon completion.
References	NK29-CORR-00531-13479 NK29-CORR-00531-12003 NK29-CORR-00531-12077 NK29-CORR-00531-09598 AI 111402

CARD #	CA-0120
CARD Title	Fuel Channel Replacement - Unit 6
CARD Description	The scope of work includes the removal and replacement of 480 fuel channels (includes end fittings, garter springs, and pressure tubes), and 480 calandria tubes. A baseline inspection in accordance with CSA N285.4 will be performed on a sample of pressure tubes.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6-Fuel Channel Replacement will have an immediate impact on maintaining the design basis of fuel channels.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of fuel channels. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-056
GIO Title	Fuel Channel Replacement
CARD(s) Associated with this GIO	CA-0120, CA-0209, CA-0226, CA-0243, CA-0260, CA-0277
Gap(s) Associated with this CARD	SF04-MCR-0001-16
Additional Information	Project Scope Document: 38826-MCR6-SoW-002-R000
References	

CARD #	CA-0209
CARD Title	Fuel Channel Replacement - Unit 3
CARD Description	The scope of work includes the removal and replacement of 480 fuel channels (includes end fittings, garter springs, and pressure tubes), and 480 calandria tubes. A baseline inspection in accordance with CSA N285.4 will be performed on a sample of pressure tubes.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3-Fuel Channel Replacement will have an immediate impact on maintaining the design basis of fuel channels.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of fuel channels. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-056
GIO Title	Fuel Channel Replacement
CARD(s) Associated with this GIO	CA-0120, CA-0209, CA-0226, CA-0243, CA-0260, CA-0277
Gap(s) Associated with this CARD	SF02-MCR-0032-16
Additional Information	
References	

CARD #	CA-0226
CARD Title	Fuel Channel Replacement - Unit 4
CARD Description	The scope of work includes the removal and replacement of 480 fuel channels (includes end fittings, garter springs, and pressure tubes), and 480 calandria tubes. A baseline inspection in accordance with CSA N285.4 will be performed on a sample of pressure tubes.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4-Fuel Channel Replacement will have an immediate impact on maintaining the design basis of fuel channels.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of fuel channels. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-056
GIO Title	Fuel Channel Replacement
CARD(s) Associated with this GIO	CA-0120, CA-0209, CA-0226, CA-0243, CA-0260, CA-0277
Gap(s) Associated with this CARD	SF02-MCR-0049-16
Additional Information	
References	

CARD #	CA-0243
CARD Title	Fuel Channel Replacement - Unit 5
CARD Description	The scope of work includes the removal and replacement of 480 fuel channels (includes end fittings, garter springs, and pressure tubes), and 480 calandria tubes. A baseline inspection in accordance with CSA N285.4 will be performed on a sample of pressure tubes.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5-Fuel Channel Replacement will have an immediate impact on maintaining the design basis of fuel channels.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of fuel channels. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-056
GIO Title	Fuel Channel Replacement
CARD(s) Associated with this GIO	CA-0120, CA-0209, CA-0226, CA-0243, CA-0260, CA-0277
Gap(s) Associated with this CARD	SF02-MCR-0066-16
Additional Information	
References	

CARD #	CA-0260
CARD Title	Fuel Channel Replacement - Unit 7
CARD Description	The scope of work includes the removal and replacement of 480 fuel channels (includes end fittings, garter springs, and pressure tubes), and 480 calandria tubes. A baseline inspection in accordance with CSA N285.4 will be performed on a sample of pressure tubes.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7-Fuel Channel Replacement will have an immediate impact on maintaining the design basis of fuel channels.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of fuel channels. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-056
GIO Title	Fuel Channel Replacement
CARD(s) Associated with this GIO	CA-0120, CA-0209, CA-0226, CA-0243, CA-0260, CA-0277
Gap(s) Associated with this CARD	SF02-MCR-0083-16
Additional Information	
References	

CARD #	CA-0277
CARD Title	Fuel Channel Replacement - Unit 8
CARD Description	The scope of work includes the removal and replacement of 480 fuel channels (includes end fittings, garter springs, and pressure tubes), and 480 calandria tubes. A baseline inspection in accordance with CSA N285.4 will be performed on a sample of pressure tubes.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8-Fuel Channel Replacement will have an immediate impact on maintaining the design basis of fuel channels.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of fuel channels. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-056
GIO Title	Fuel Channel Replacement
CARD(s) Associated with this GIO	CA-0120, CA-0209, CA-0226, CA-0243, CA-0260, CA-0277
Gap(s) Associated with this CARD	SF02-MCR-0100-16
Additional Information	
References	

CARD #	CA-0121
CARD Title	Steam Generator Replacement - Unit 6
CARD Description	The scope of work includes replacement of all 8 steam generator cartridges with new as well as support stool assemblies. Steam Generator cartridge (lower portion) contains the primary head, tubesheet and tubing and the lower secondary shell. The upper portion contains the steam drum (i.e. the steam separation equipment). The steam drum will undergo inspection, repair and refurbishment and will be reattached to the replacement steam generator cartridge. The scope of work also includes inspection, testing, repair and maintenance of steam generator containment bellows assembly, seal, plates, seismic restraints. Inaugural inspection of the entire assembly is performed in accordance with CSA N285.4 upon completion of the installation.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6-Steam Generator Replacement will have an immediate impact on maintaining the design basis of Steam Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Steam Generators. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-057
GIO Title	Steam Generator Replacement
CARD(s) Associated with this GIO	CA-0121, CA-0210, CA-0227, CA-0244, CA-0261, CA-0278
Gap(s) Associated with this CARD	SF04-MCR-0002-16
Additional Information	Project Scope Document: 38827-MCR6-SoW-002-R000

References	

CARD #	CA-0210
CARD Title	Steam Generator Replacement - Unit 3
CARD Description	The scope of work includes replacement of all 8 steam generator cartridges with new as well as support stool assemblies. Steam Generator cartridge (lower portion) contains the primary head, tubesheet and tubing and the lower secondary shell. The upper portion contains the steam drum (i.e. the steam separation equipment). The steam drum will undergo inspection, repair and refurbishment and will be reattached to the replacement steam generator cartridge. The scope of work also includes inspection, testing, repair and maintenance of steam generator containment bellows assembly, seal, plates, seismic restraints. Inaugural inspection of the entire assembly is performed in accordance with CSA N285.4 upon completion of the installation.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3-Steam Generator Replacement will have an immediate impact on maintaining the design basis of Steam Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Steam Generators. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-057
GIO Title	Steam Generator Replacement
CARD(s) Associated with this GIO	CA-0121, CA-0210, CA-0227, CA-0244, CA-0261, CA-0278
Gap(s) Associated with this CARD	SF02-MCR-0033-16
Additional Information	AMOT-0007.

References	

CARD #	CA-0227
CARD Title	Steam Generator Replacement - Unit 4
CARD Description	The scope of work includes replacement of all 8 steam generator cartridges with new as well as support stool assemblies. Steam Generator cartridge (lower portion) contains the primary head, tubesheet and tubing and the lower secondary shell. The upper portion contains the steam drum (i.e. the steam separation equipment). The steam drum will undergo inspection, repair and refurbishment and will be reattached to the replacement steam generator cartridge. The scope of work also includes inspection, testing, repair and maintenance of steam generator containment bellows assembly, seal, plates, seismic restraints. Inaugural inspection of the entire assembly is performed in accordance with CSA N285.4 upon completion of the installation.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4-Steam Generator Replacement will have an immediate impact on maintaining the design basis of Steam Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Steam Generators. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-057
GIO Title	Steam Generator Replacement
CARD(s) Associated with this GIO	CA-0121, CA-0210, CA-0227, CA-0244, CA-0261, CA-0278
Gap(s) Associated with this CARD	SF02-MCR-0050-16
Additional Information	AMOT-0007.

References	

CARD #	CA-0244
CARD Title	Steam Generator Replacement - Unit 5
CARD Description	The scope of work includes replacement of all 8 steam generator cartridges with new as well as support stool assemblies. Steam Generator cartridge (lower portion) contains the primary head, tubesheet and tubing and the lower secondary shell. The upper portion contains the steam drum (i.e. the steam separation equipment). The steam drum will undergo inspection, repair and refurbishment and will be reattached to the replacement steam generator cartridge. The scope of work also includes inspection, testing, repair and maintenance of steam generator containment bellows assembly, seal, plates, seismic restraints. Inaugural inspection of the entire assembly is performed in accordance with CSA N285.4 upon completion of the installation.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5-Steam Generator Replacement will have an immediate impact on maintaining the design basis of Steam Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Steam Generators. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-057
GIO Title	Steam Generator Replacement
CARD(s) Associated with this GIO	CA-0121, CA-0210, CA-0227, CA-0244, CA-0261, CA-0278
Gap(s) Associated with this CARD	SF02-MCR-0067-16
Additional Information	AMOT-0010.

References	

CARD #	CA-0261
CARD Title	Steam Generator Replacement - Unit 7
CARD Description	The scope of work includes replacement of all 8 steam generator cartridges with new as well as support stool assemblies. Steam Generator cartridge (lower portion) contains the primary head, tubesheet and tubing and the lower secondary shell. The upper portion contains the steam drum (i.e. the steam separation equipment). The steam drum will undergo inspection, repair and refurbishment and will be reattached to the replacement steam generator cartridge. The scope of work also includes inspection, testing, repair and maintenance of steam generator containment bellows assembly, seal, plates, seismic restraints. Inaugural inspection of the entire assembly is performed in accordance with CSA N285.4 upon completion of the installation.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7-Steam Generator Replacement will have an immediate impact on maintaining the design basis of Steam Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Steam Generators. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-057
GIO Title	Steam Generator Replacement
CARD(s) Associated with this GIO	CA-0121, CA-0210, CA-0227, CA-0244, CA-0261, CA-0278
Gap(s) Associated with this CARD	SF02-MCR-0084-16
Additional Information	AMOT-0010.

References	

CARD #	CA-0278
CARD Title	Steam Generator Replacement - Unit 8
CARD Description	The scope of work includes replacement of all 8 steam generator cartridges with new as well as support stool assemblies. Steam Generator cartridge (lower portion) contains the primary head, tubesheet and tubing and the lower secondary shell. The upper portion contains the steam drum (i.e. the steam separation equipment). The steam drum will undergo inspection, repair and refurbishment and will be reattached to the replacement steam generator cartridge. The scope of work also includes inspection, testing, repair and maintenance of steam generator containment bellows assembly, seal, plates, seismic restraints. Inaugural inspection of the entire assembly is performed in accordance with CSA N285.4 upon completion of the installation.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8-Steam Generator Replacement will have an immediate impact on maintaining the design basis of Steam Generators.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Steam Generators. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-057
GIO Title	Steam Generator Replacement
CARD(s) Associated with this GIO	CA-0121, CA-0210, CA-0227, CA-0244, CA-0261, CA-0278
Gap(s) Associated with this CARD	SF02-MCR-0101-16
Additional Information	AMOT-0010.

References	

CARD #	CA-0122
CARD Title	Feeder Replacement - Unit 6
CARD Description	The scope of work includes replacement of all 960 inlet and outlet feeders and associated supports, instruments lines/guide tubes, remote temperature detectors (RTD) and cabling. A baseline inspection in accordance with CSA N285.4 is performed upon completion of the installation of new components.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6-Feeder Replacement will have an immediate impact on maintaining the design basis of Feeders.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Feeders. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-058
GIO Title	Feeder Replacement
CARD(s) Associated with this GIO	CA-0122, CA-0211, CA-0228, CA-0245, CA-0262, CA-0279
Gap(s) Associated with this CARD	SF04-MCR-0003-16
Additional Information	38828-MCR6-SoW-002-R001.
References	

CARD #	CA-0211
CARD Title	Feeder Replacement - Unit 3
CARD Description	The scope of work includes replacement of all 960 inlet and outlet feeders and associated supports, instruments lines/guide tubes, remote temperature detectors (RTD) and cabling. A baseline inspection in accordance with CSA N285.4 is performed upon completion of the installation of new components.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3-Feeder Replacement will have an immediate impact on maintaining the design basis of Feeders.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Feeders. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-058
GIO Title	Feeder Replacement
CARD(s) Associated with this GIO	CA-0122, CA-0211, CA-0228, CA-0245, CA-0262, CA-0279
Gap(s) Associated with this CARD	SF02-MCR-0034-16
Additional Information	
References	

CARD #	CA-0228
CARD Title	Feeder Replacement - Unit 4
CARD Description	The scope of work includes replacement of all 960 inlet and outlet feeders and associated supports, instruments lines/guide tubes, remote temperature detectors (RTD) and cabling. A baseline inspection in accordance with CSA N285.4 is performed upon completion of the installation of new components.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4-Feeder Replacement will have an immediate impact on maintaining the design basis of Feeders.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Feeders. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-058
GIO Title	Feeder Replacement
CARD(s) Associated with this GIO	CA-0122, CA-0211, CA-0228, CA-0245, CA-0262, CA-0279
Gap(s) Associated with this CARD	SF02-MCR-0051-16
Additional Information	
References	

CARD #	CA-0245
CARD Title	Feeder Replacement - Unit 5
CARD Description	The scope of work includes replacement of all 960 inlet and outlet feeders and associated supports, instruments lines/guide tubes, remote temperature detectors (RTD) and cabling. A baseline inspection in accordance with CSA N285.4 is performed upon completion of the installation of new components.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5-Feeder Replacement will have an immediate impact on maintaining the design basis of Feeders.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Feeders. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-058
GIO Title	Feeder Replacement
CARD(s) Associated with this GIO	CA-0122, CA-0211, CA-0228, CA-0245, CA-0262, CA-0279
Gap(s) Associated with this CARD	SF02-MCR-0068-16
Additional Information	
References	

CARD #	CA-0262
CARD Title	Feeder Replacement - Unit 7
CARD Description	The scope of work includes replacement of all 960 inlet and outlet feeders and associated supports, instruments lines/guide tubes, remote temperature detectors (RTD) and cabling. A baseline inspection in accordance with CSA N285.4 is performed upon completion of the installation of new components.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7-Feeder Replacement will have an immediate impact on maintaining the design basis of Feeders.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Feeders. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-058
GIO Title	Feeder Replacement
CARD(s) Associated with this GIO	CA-0122, CA-0211, CA-0228, CA-0245, CA-0262, CA-0279
Gap(s) Associated with this CARD	SF02-MCR-0085-16
Additional Information	
References	

CARD #	CA-0279
CARD Title	Feeder Replacement - Unit 8
CARD Description	The scope of work includes replacement of all 960 inlet and outlet feeders and associated supports, instruments lines/guide tubes, remote temperature detectors (RTD) and cabling. A baseline inspection in accordance with CSA N285.4 is performed upon completion of the installation of new components.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8-Feeder Replacement will have an immediate impact on maintaining the design basis of Feeders.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Feeders. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-058
GIO Title	Feeder Replacement
CARD(s) Associated with this GIO	CA-0122, CA-0211, CA-0228, CA-0245, CA-0262, CA-0279
Gap(s) Associated with this CARD	SF02-MCR-0102-16
Additional Information	
References	

CARD #	CA-0126
CARD Title	PHT Pump Seal Bellows Replacement - Unit 6
CARD Description	The scope of work includes: - Inspection of four in-situ pump seal bellow assemblies - Preparation of technical basis for extended life of each bellows seal and securing approval for life extension - Replacement of seal bellows that are not approved for extended life - Post inspection and testing of replaced seal bellows
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- PHT Pump Seal Bellows Replacement will have an immediate impact on maintaining the design basis of PHT Pumps.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of PHT Pumps. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-062
GIO Title	PHT Pump Seal Bellows Replacement
CARD(s) Associated with this GIO	CA-0126, CA-0215, CA-0232, CA-0249, CA-0266, CA-0283
Gap(s) Associated with this CARD	SF04-MCR-0007-16
Additional Information	38842-MCR6-SoW-002
References	

CARD #	CA-0215
CARD Title	PHT Pump Seal Bellows Replacement - Unit 3
CARD Description	The scope of work includes: - Inspection of four in-situ pump seal bellow assemblies - Preparation of technical basis for extended life of each bellows seal and securing approval for life extension - Replacement of seal bellows that are not approved for extended life - Post inspection and testing of replaced seal bellows
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- PHT Pump Seal Bellows Replacement will have an immediate impact on maintaining the design basis of PHT Pumps.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of PHT Pumps. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-062
GIO Title	PHT Pump Seal Bellows Replacement
CARD(s) Associated with this GIO	CA-0126, CA-0215, CA-0232, CA-0249, CA-0266, CA-0283
Gap(s) Associated with this CARD	SF02-MCR-0038-16
Additional Information	
References	

CARD #	CA-0232
CARD Title	PHT Pump Seal Bellows Replacement - Unit 4
CARD Description	The scope of work includes: - Inspection of four in-situ pump seal bellow assemblies - Preparation of technical basis for extended life of each bellows seal and securing approval for life extension - Replacement of seal bellows that are not approved for extended life - Post inspection and testing of replaced seal bellows
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- PHT Pump Seal Bellows Replacement will have an immediate impact on maintaining the design basis of PHT Pumps.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of PHT Pumps. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-062
GIO Title	PHT Pump Seal Bellows Replacement
CARD(s) Associated with this GIO	CA-0126, CA-0215, CA-0232, CA-0249, CA-0266, CA-0283
Gap(s) Associated with this CARD	SF02-MCR-0055-16
Additional Information	
References	

CARD #	CA-0249
CARD Title	PHT Pump Seal Bellows Replacement - Unit 5
CARD Description	The scope of work includes: - Inspection of four in-situ pump seal bellow assemblies - Preparation of technical basis for extended life of each bellows seal and securing approval for life extension - Replacement of seal bellows that are not approved for extended life - Post inspection and testing of replaced seal bellows
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- PHT Pump Seal Bellows Replacement will have an immediate impact on maintaining the design basis of PHT Pumps.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of PHT Pumps. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-062
GIO Title	PHT Pump Seal Bellows Replacement
CARD(s) Associated with this GIO	CA-0126, CA-0215, CA-0232, CA-0249, CA-0266, CA-0283
Gap(s) Associated with this CARD	SF02-MCR-0072-16
Additional Information	
References	

CARD #	CA-0266
CARD Title	PHT Pump Seal Bellows Replacement - Unit 7
CARD Description	The scope of work includes: - Inspection of four in-situ pump seal bellow assemblies - Preparation of technical basis for extended life of each bellows seal and securing approval for life extension - Replacement of seal bellows that are not approved for extended life - Post inspection and testing of replaced seal bellows
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- PHT Pump Seal Bellows Replacement will have an immediate impact on maintaining the design basis of PHT Pumps.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of PHT Pumps. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-062
GIO Title	PHT Pump Seal Bellows Replacement
CARD(s) Associated with this GIO	CA-0126, CA-0215, CA-0232, CA-0249, CA-0266, CA-0283
Gap(s) Associated with this CARD	SF02-MCR-0089-16
Additional Information	
References	

CARD #	CA-0283
CARD Title	PHT Pump Seal Bellows Replacement - Unit 8
CARD Description	The scope of work includes: - Inspection of four in-situ pump seal bellow assemblies - Preparation of technical basis for extended life of each bellows seal and securing approval for life extension - Replacement of seal bellows that are not approved for extended life - Post inspection and testing of replaced seal bellows
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- PHT Pump Seal Bellows Replacement will have an immediate impact on maintaining the design basis of PHT Pumps.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of PHT Pumps. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-062
GIO Title	PHT Pump Seal Bellows Replacement
CARD(s) Associated with this GIO	CA-0126, CA-0215, CA-0232, CA-0249, CA-0266, CA-0283
Gap(s) Associated with this CARD	SF02-MCR-0106-16
Additional Information	
References	

CARD #	CA-0138
CARD Title	Air Operated Valve- Nuclear Valve Replacement - Unit 6
CARD Description	The scope of work includes complete replacement of entire valve/actuator assemblies of critical Bruce B HTS Air Operated Valves: - 33310- CV11, CV12, CV14 - 33320- CV5, CV6, CV20, CV21, CV22, CV23
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Air Operated Valve- Nuclear Valve Replacement have an immediate impact on maintaining the design basis of Air Operated Nuclear Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Air Operated Nuclear Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF04-MCR-0019-16
Additional Information	AMOT-0345
References	

CARD #	CA-0139
CARD Title	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 6
CARD Description	The scope of work covers the replacement of 25 Newman Hattersley Bellow Sealed valves installed in the Heat Transport and ECI systems. These are originally installed equipment, most of which have no previous maintenance. - 33310-MV20 - 33320-CV46, CV47, MV3, -MV9, MV30, MV35, MV37 - 33330-MV3, MV12, MV15, MV27 - 33340-MV31, MV32, MV33 - 33810-MV1 - 34330-MV105, MV106, MV108, MV109, MV112, MV113, MV114 - 34720-MV15, MV16
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Air Operated Valve- Newman Hattersley (N/H) Bellow Sealed Valves Replacement will have an immediate impact on maintaining the design basis of (N/H) Bellow Sealed Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of (N/H) Bellow Sealed Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF04-MCR-0020-16

Additional Information	AMOT-0347
References	

CARD #	CA-0217
CARD Title	Air Operated Valve- Nuclear Valve Replacement - Unit 3
CARD Description	The scope of work includes complete replacement of entire valve/actuator assemblies of critical Bruce A HTS Air Operated Valves.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Air Operated Valve- Nuclear Valve Replacement have an immediate impact on maintaining the design basis of Air Operated Nuclear Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Air Operated Nuclear Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0040-16
Additional Information	AMOT-0344
References	

CARD #	CA-0234
CARD Title	Air Operated Valve- Nuclear Valve Replacement - Unit 4
CARD Description	The scope of work includes complete replacement of entire valve/actuator assemblies of critical Bruce A HTS Air Operated Valves.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Air Operated Valve- Nuclear Valve Replacement have an immediate impact on maintaining the design basis of Air Operated Nuclear Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Air Operated Nuclear Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0057-16
Additional Information	AMOT-0344
References	

CARD #	CA-0251
CARD Title	Air Operated Valve- Nuclear Valve Replacement - Unit 5
CARD Description	The scope of work includes complete replacement of entire valve/actuator assemblies of critical Bruce B HTS Air Operated Valves.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Air Operated Valve- Nuclear Valve Replacement have an immediate impact on maintaining the design basis of Air Operated Nuclear Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Air Operated Nuclear Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0074-16
Additional Information	AMOT-0345
References	

CARD #	CA-0268
CARD Title	Air Operated Valve- Nuclear Valve Replacement - Unit 7
CARD Description	The scope of work includes complete replacement of entire valve/actuator assemblies of critical Bruce B HTS Air Operated Valves.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Air Operated Valve- Nuclear Valve Replacement have an immediate impact on maintaining the design basis of Air Operated Nuclear Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Air Operated Nuclear Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0091-16
Additional Information	AMOT-0345
References	

CARD #	CA-0285
CARD Title	Air Operated Valve- Nuclear Valve Replacement - Unit 8
CARD Description	The scope of work includes complete replacement of entire valve/actuator assemblies of critical Bruce B HTS Air Operated Valves.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Air Operated Valve- Nuclear Valve Replacement have an immediate impact on maintaining the design basis of Air Operated Nuclear Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Air Operated Nuclear Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0108-16
Additional Information	AMOT-0345
References	

CARD #	CA-0329
CARD Title	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 3
CARD Description	The scope of work covers the replacement of Newman Hattersley Bellow Sealed valves installed in the Heat Transport and ECI systems. These are originally installed equipment, most of which have no previous maintenance.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Air Operated Valve- Newman Hattersley (N/H) Bellow Sealed Valves Replacement will have an immediate impact on maintaining the design basis of (N/H) Bellow Sealed Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of (N/H) Bellow Sealed Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0125-16
Additional Information	AMOT-0346
References	

CARD #	CA-0330
CARD Title	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 4
CARD Description	The scope of work covers the replacement of Newman Hattersley Bellow Sealed valves installed in the Heat Transport and ECI systems. These are originally installed equipment, most of which have no previous maintenance.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Air Operated Valve- Newman Hattersley (N/H) Bellow Sealed Valves Replacement will have an immediate impact on maintaining the design basis of (N/H) Bellow Sealed Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of (N/H) Bellow Sealed Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0126-16
Additional Information	AMOT-0346
References	

CARD #	CA-0331
CARD Title	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 5
CARD Description	The scope of work covers the replacement of Newman Hattersley Bellow Sealed valves installed in the Heat Transport and ECI systems. These are originally installed equipment, most of which have no previous maintenance.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Air Operated Valve- Newman Hattersley (N/H) Bellow Sealed Valves Replacement will have an immediate impact on maintaining the design basis of (N/H) Bellow Sealed Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of (N/H) Bellow Sealed Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0127-16
Additional Information	AMOT-0347
References	

CARD #	CA-0332
CARD Title	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 7
CARD Description	The scope of work covers the replacement of Newman Hattersley Bellow Sealed valves installed in the Heat Transport and ECI systems. These are originally installed equipment, most of which have no previous maintenance.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Air Operated Valve- Newman Hattersley (N/H) Bellow Sealed Valves Replacement will have an immediate impact on maintaining the design basis of (N/H) Bellow Sealed Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of (N/H) Bellow Sealed Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0128-16
Additional Information	AMOT-0347
References	

CARD #	CA-0333
CARD Title	Air Operated Valve - Newman Hattersley (N/H) Bellow Sealed Valves - Unit 8
CARD Description	The scope of work covers the replacement of Newman Hattersley Bellow Sealed valves installed in the Heat Transport and ECI systems. These are originally installed equipment, most of which have no previous maintenance.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Air Operated Valve- Newman Hattersley (N/H) Bellow Sealed Valves Replacement will have an immediate impact on maintaining the design basis of (N/H) Bellow Sealed Valves.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of (N/H) Bellow Sealed Valves. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-070
GIO Title	Air Operated Valves-Replacement
CARD(s) Associated with this GIO	CA-0138, CA-0139, CA-0217, CA-0234, CA-0251, CA-0268, CA-0285, CA-0329, CA-0330, CA-0331, CA-0332, CA-0333
Gap(s) Associated with this CARD	SF02-MCR-0129-16
Additional Information	AMOT-0347
References	

CARD #	CA-0145
CARD Title	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 6
CARD Description	Restore the reliability of the Maintenance Cooling pump motors 34720-PM1 and PM2. This will be accomplished by purchasing two new replacement motors and cycling these through the units and refurbish each motor. At the end of the project the remaining motor(s) will be refurbished as viable spares or scrapped.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Maintenance Cooling System (MCS) Pump Motor Refurbishment/Replacement will have an immediate impact on maintaining the design basis of MCS Pump Motors.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of MCS Pump Motors. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-071
GIO Title	Large Motors-Refurbishment/Replacement
CARD(s) Associated with this GIO	CA-0145, CA-0346, CA-0347, CA-0348, CA-0352, CA-0353
Gap(s) Associated with this CARD	SF04-MCR-0026-16
Additional Information	AMOT-0285
References	

CARD #	CA-0346
CARD Title	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 3
CARD Description	Restore the reliability of the Maintenance Cooling pump motors 34720-PM1 and PM2. This will be accomplished by purchasing two new replacement motors and cycling these through the units and refurbish each motor. At the end of the project the remaining motor(s) will be refurbished as viable spares or scrapped.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Maintenance Cooling System (MCS) Pump Motor Refurbishment/Replacement will have an immediate impact on maintaining the design basis of MCS Pump Motors.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of MCS Pump Motors. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-071
GIO Title	Large Motors-Refurbishment/Replacement
CARD(s) Associated with this GIO	CA-0145, CA-0346, CA-0347, CA-0348, CA-0352, CA-0353
Gap(s) Associated with this CARD	SF02-MCR-0161-16
Additional Information	AMOT-0283
References	

CARD #	CA-0347
CARD Title	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 4
CARD Description	Restore the reliability of the Maintenance Cooling pump motors 34720-PM1 and PM2. This will be accomplished by purchasing two new replacement motors and cycling these through the units and refurbish each motor. At the end of the project the remaining motor(s) will be refurbished as viable spares or scrapped.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Maintenance Cooling System (MCS) Pump Motor Refurbishment/Replacement will have an immediate impact on maintaining the design basis of MCS Pump Motors.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of MCS Pump Motors. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-071
GIO Title	Large Motors-Refurbishment/Replacement
CARD(s) Associated with this GIO	CA-0145, CA-0346, CA-0347, CA-0348, CA-0352, CA-0353
Gap(s) Associated with this CARD	SF02-MCR-0162-16
Additional Information	AMOT-0283
References	

CARD #	CA-0348
CARD Title	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 5
CARD Description	Restore the reliability of the Maintenance Cooling pump motors 34720-PM1 and PM2. This will be accomplished by purchasing two new replacement motors and cycling these through the units and refurbish each motor. At the end of the project the remaining motor(s) will be refurbished as viable spares or scrapped.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Maintenance Cooling System (MCS) Pump Motor Refurbishment/Replacement will have an immediate impact on maintaining the design basis of MCS Pump Motors.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of MCS Pump Motors. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-071
GIO Title	Large Motors-Refurbishment/Replacement
CARD(s) Associated with this GIO	CA-0145, CA-0346, CA-0347, CA-0348, CA-0352, CA-0353
Gap(s) Associated with this CARD	SF02-MCR-0163-16
Additional Information	AMOT-0285
References	

CARD #	CA-0352
CARD Title	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 7
CARD Description	Restore the reliability of the Maintenance Cooling pump motors 34720-PM1 and PM2. This will be accomplished by purchasing two new replacement motors and cycling these through the units and refurbish each motor. At the end of the project the remaining motor(s) will be refurbished as viable spares or scrapped.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Maintenance Cooling System (MCS) Pump Motor Refurbishment/Replacement will have an immediate impact on maintaining the design basis of MCS Pump Motors.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of MCS Pump Motors. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-071
GIO Title	Large Motors-Refurbishment/Replacement
CARD(s) Associated with this GIO	CA-0145, CA-0346, CA-0347, CA-0348, CA-0352, CA-0353
Gap(s) Associated with this CARD	SF02-MCR-0167-16
Additional Information	AMOT-0285
References	

CARD #	CA-0353
CARD Title	Large Motors - Maintenance Cooling System (MCS) Pump Motors Refurbishment/Replacement - Unit 8
CARD Description	Restore the reliability of the Maintenance Cooling pump motors 34720-PM1 and PM2. This will be accomplished by purchasing two new replacement motors and cycling these through the units and refurbish each motor. At the end of the project the remaining motor(s) will be refurbished as viable spares or scrapped.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Maintenance Cooling System (MCS) Pump Motor Refurbishment/Replacement will have an immediate impact on maintaining the design basis of MCS Pump Motors.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of MCS Pump Motors. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-071
GIO Title	Large Motors-Refurbishment/Replacement
CARD(s) Associated with this GIO	CA-0145, CA-0346, CA-0347, CA-0348, CA-0352, CA-0353
Gap(s) Associated with this CARD	SF02-MCR-0168-16
Additional Information	AMOT-0285
References	

CARD #	CA-0153
CARD Title	DCC Cables and WIBAs -Replacement - Unit 6
CARD Description	This scope includes the replacement of the existing interconnect cables and I/O cables/ Weidmeuller Interface Boards (WIBAs). The interconnect cables run between chassis and ultimately back to the computer I/O bus. The I/O and WIBA cables run from the I/O chassis to the WIBA assemblies.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- DCC Cables and WIBAs Replacement will have an immediate impact on maintaining the design basis of DCCs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of DCCs. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-076
GIO Title	DCC Cables and WIBAs –Replacement
CARD(s) Associated with this GIO	CA-0153, CA-0221, CA-0238, CA-0255, CA-0272, CA-0289
Gap(s) Associated with this CARD	SF04-MCR-0034-16
Additional Information	AMOT-0034
References	

CARD #	CA-0221
CARD Title	DCC Cables and WIBAs -Replacement - Unit 3
CARD Description	This scope includes the replacement of the existing interconnect cables and I/O cables/Weidmeuller Interface Boards (WIBAs). The interconnect cables run between chassis and ultimately back to the computer I/O bus. The I/O and WIBA cables run from the I/O chassis to the WIBA assemblies.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- DCC Cables and WIBAs Replacement will have an immediate impact on maintaining the design basis of DCCs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of DCCs. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-076
GIO Title	DCC Cables and WIBAs –Replacement
CARD(s) Associated with this GIO	CA-0153, CA-0221, CA-0238, CA-0255, CA-0272, CA-0289
Gap(s) Associated with this CARD	SF02-MCR-0044-16
Additional Information	AMOT-0033
References	

CARD #	CA-0238
CARD Title	DCC Cables and WIBAs -Replacement - Unit 4
CARD Description	This scope includes the replacement of the existing interconnect cables and I/O cables/Weidmeuller Interface Boards (WIBAs). The interconnect cables run between chassis and ultimately back to the computer I/O bus. The I/O and WIBA cables run from the I/O chassis to the WIBA assemblies.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- DCC Cables and WIBAs Replacement will have an immediate impact on maintaining the design basis of DCCs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of DCCs. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-076
GIO Title	DCC Cables and WIBAs –Replacement
CARD(s) Associated with this GIO	CA-0153, CA-0221, CA-0238, CA-0255, CA-0272, CA-0289
Gap(s) Associated with this CARD	SF02-MCR-0061-16
Additional Information	AMOT-0033
References	

CARD #	CA-0255
CARD Title	DCC Cables and WIBAs -Replacement - Unit 5
CARD Description	This scope includes the replacement of the existing interconnect cables and I/O cables/Weidmeuller Interface Boards (WIBAs). The interconnect cables run between chassis and ultimately back to the computer I/O bus. The I/O and WIBA cables run from the I/O chassis to the WIBA assemblies.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- DCC Cables and WIBAs Replacement will have an immediate impact on maintaining the design basis of DCCs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of DCCs. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-076
GIO Title	DCC Cables and WIBAs –Replacement
CARD(s) Associated with this GIO	CA-0153, CA-0221, CA-0238, CA-0255, CA-0272, CA-0289
Gap(s) Associated with this CARD	SF02-MCR-0078-16
Additional Information	AMOT-0034
References	

CARD #	CA-0272
CARD Title	DCC Cables and WIBAs -Replacement - Unit 7
CARD Description	This scope includes the replacement of the existing interconnect cables and I/O cables/Weidmeuller Interface Boards (WIBAs). The interconnect cables run between chassis and ultimately back to the computer I/O bus. The I/O and WIBA cables run from the I/O chassis to the WIBA assemblies.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- DCC Cables and WIBAs Replacement will have an immediate impact on maintaining the design basis of DCCs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of DCCs. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-076
GIO Title	DCC Cables and WIBAs –Replacement
CARD(s) Associated with this GIO	CA-0153, CA-0221, CA-0238, CA-0255, CA-0272, CA-0289
Gap(s) Associated with this CARD	SF02-MCR-0095-16
Additional Information	AMOT-0034
References	

CARD #	CA-0289
CARD Title	DCC Cables and WIBAs -Replacement - Unit 8
CARD Description	This scope includes the replacement of the existing interconnect cables and I/O cables/Weidmeuller Interface Boards (WIBAs). The interconnect cables run between chassis and ultimately back to the computer I/O bus. The I/O and WIBA cables run from the I/O chassis to the WIBA assemblies.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- DCC Cables and WIBAs Replacement will have an immediate impact on maintaining the design basis of DCCs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of DCCs. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-076
GIO Title	DCC Cables and WIBAs –Replacement
CARD(s) Associated with this GIO	CA-0153, CA-0221, CA-0238, CA-0255, CA-0272, CA-0289
Gap(s) Associated with this CARD	SF02-MCR-0112-16
Additional Information	AMOT-0034
References	

CARD #	CA-0154
CARD Title	Moderator Heat Exchangers- Replacement - Unit 6
CARD Description	The scope of work covers replacement of Moderator Heat Exchangers 32110-HX1 and HX2
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Moderator Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Moderator Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Moderator Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	23-Dec-33
GIO #	GIO-077
GIO Title	Moderator Heat Exchangers- Replacement
CARD(s) Associated with this GIO	CA-0154, CA-0222, CA-0239, CA-0256, CA-0273, CA-0290
Gap(s) Associated with this CARD	SF04-MCR-0035-16
Additional Information	AMOT-0044
References	

CARD #	CA-0222
CARD Title	Moderator Heat Exchangers- Replacement - Unit 3
CARD Description	The scope of work covers replacement of Moderator Heat Exchangers.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Moderator Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Moderator Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Moderator Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-077
GIO Title	Moderator Heat Exchangers- Replacement
CARD(s) Associated with this GIO	CA-0154, CA-0222, CA-0239, CA-0256, CA-0273, CA-0290
Gap(s) Associated with this CARD	SF02-MCR-0045-16
Additional Information	AMOT-0043
References	

CARD #	CA-0239
CARD Title	Moderator Heat Exchangers- Replacement - Unit 4
CARD Description	The scope of work covers replacement of Moderator Heat Exchangers.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Moderator Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Moderator Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Moderator Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-077
GIO Title	Moderator Heat Exchangers- Replacement
CARD(s) Associated with this GIO	CA-0154, CA-0222, CA-0239, CA-0256, CA-0273, CA-0290
Gap(s) Associated with this CARD	SF02-MCR-0062-16
Additional Information	AMOT-0043
References	

CARD #	CA-0256
CARD Title	Moderator Heat Exchangers- Replacement - Unit 5
CARD Description	The scope of work covers replacement of Moderator Heat Exchangers.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Moderator Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Moderator Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Moderator Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-077
GIO Title	Moderator Heat Exchangers- Replacement
CARD(s) Associated with this GIO	CA-0154, CA-0222, CA-0239, CA-0256, CA-0273, CA-0290
Gap(s) Associated with this CARD	SF02-MCR-0079-16
Additional Information	AMOT-0044
References	

CARD #	CA-0273
CARD Title	Moderator Heat Exchangers- Replacement - Unit 7
CARD Description	The scope of work covers replacement of Moderator Heat Exchangers.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Moderator Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Moderator Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Moderator Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-077
GIO Title	Moderator Heat Exchangers- Replacement
CARD(s) Associated with this GIO	CA-0154, CA-0222, CA-0239, CA-0256, CA-0273, CA-0290
Gap(s) Associated with this CARD	SF02-MCR-0096-16
Additional Information	AMOT-0044
References	

CARD #	CA-0290
CARD Title	Moderator Heat Exchangers- Replacement - Unit 8
CARD Description	The scope of work covers replacement of Moderator Heat Exchangers.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Moderator Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Moderator Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Moderator Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-077
GIO Title	Moderator Heat Exchangers- Replacement
CARD(s) Associated with this GIO	CA-0154, CA-0222, CA-0239, CA-0256, CA-0273, CA-0290
Gap(s) Associated with this CARD	SF02-MCR-0113-16
Additional Information	AMOT-0044
References	

CARD #	CA-0155
CARD Title	Maintenance Cooling Heat Exchanger- Replacement - Unit 6
CARD Description	The scope of work covers replacement of Maintenance Cooling Heat Exchanger 34720-HX1
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Maintenance Cooling Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Maintenance Cooling Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-078
GIO Title	Maintenance Cooling Heat Exchanger- Replacement
CARD(s) Associated with this GIO	CA-0155, CA-0223, CA-0240, CA-0257, CA-0274, CA-0291
Gap(s) Associated with this CARD	SF04-MCR-0036-16
Additional Information	AMOT-0046
References	

CARD #	CA-0223
CARD Title	Maintenance Cooling Heat Exchanger- Replacement - Unit 3
CARD Description	The scope of work covers replacement of Maintenance Cooling Heat Exchanger.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Maintenance Cooling Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Maintenance Cooling Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-078
GIO Title	Maintenance Cooling Heat Exchanger- Replacement
CARD(s) Associated with this GIO	CA-0155, CA-0223, CA-0240, CA-0257, CA-0274, CA-0291
Gap(s) Associated with this CARD	SF02-MCR-0046-16
Additional Information	AMOT-0045
References	

CARD #	CA-0240
CARD Title	Maintenance Cooling Heat Exchanger- Replacement - Unit 4
CARD Description	The scope of work covers replacement of Maintenance Cooling Heat Exchanger.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Maintenance Cooling Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Maintenance Cooling Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-078
GIO Title	Maintenance Cooling Heat Exchanger- Replacement
CARD(s) Associated with this GIO	CA-0155, CA-0223, CA-0240, CA-0257, CA-0274, CA-0291
Gap(s) Associated with this CARD	SF02-MCR-0063-16
Additional Information	AMOT-0045
References	

CARD #	CA-0257
CARD Title	Maintenance Cooling Heat Exchanger- Replacement - Unit 5
CARD Description	The scope of work covers replacement of Maintenance Cooling Heat Exchanger.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Maintenance Cooling Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Maintenance Cooling Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-078
GIO Title	Maintenance Cooling Heat Exchanger- Replacement
CARD(s) Associated with this GIO	CA-0155, CA-0223, CA-0240, CA-0257, CA-0274, CA-0291
Gap(s) Associated with this CARD	SF02-MCR-0080-16
Additional Information	AMOT-0046
References	

CARD #	CA-0274
CARD Title	Maintenance Cooling Heat Exchanger- Replacement - Unit 7
CARD Description	The scope of work covers replacement of Maintenance Cooling Heat Exchanger.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Maintenance Cooling Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Maintenance Cooling Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-078
GIO Title	Maintenance Cooling Heat Exchanger- Replacement
CARD(s) Associated with this GIO	CA-0155, CA-0223, CA-0240, CA-0257, CA-0274, CA-0291
Gap(s) Associated with this CARD	SF02-MCR-0097-16
Additional Information	AMOT-0046
References	

CARD #	CA-0291
CARD Title	Maintenance Cooling Heat Exchanger- Replacement - Unit 8
CARD Description	The scope of work covers replacement of Maintenance Cooling Heat Exchanger.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Maintenance Cooling Heat Exchanger Replacement will have an immediate impact on maintaining the design basis of Maintenance Cooling Heat Exchangers.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling Heat Exchangers. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-078
GIO Title	Maintenance Cooling Heat Exchanger- Replacement
CARD(s) Associated with this GIO	CA-0155, CA-0223, CA-0240, CA-0257, CA-0274, CA-0291
Gap(s) Associated with this CARD	SF02-MCR-0114-16
Additional Information	AMOT-0046
References	

CARD #	CA-0354
CARD Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 3
CARD Description	The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 3 includes the following physical modifications: - Replacement of pressure regulating valve PRV28 - Installation of a new relief valve RV80 on L27, set at 220 psig, downstream of PRV28 - Installation of a new pressure regulating valve PRV59 on line L34, upstream of PRV44 (simply used to limit the maximum flow through PRV44 in case it ever fails open) - Replacement of RV45 on line L34 with a model that is ASME code-compliant - Installation of a new relief valve RV511 on line L33, set at 50 psig, downstream of PRV49 (to protect the interfacing PHTS D2O Storage, Transfer and Recovery system from overpressure) The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 3 includes the following non-physical modifications: - Increase the design pressure of various Class 6 portions of the Moderator Cover Gas system that directly interface with the Bulk Helium Supply system, from 150 psig to 220 psig, to match the design pressure of the Bulk Helium Supply system (technical justification based on NK21-CALC-32310-00007 and NK21-CALC-32310-00008. - Lower the design pressure of various Class 6 portions of the Moderator Cover Gas system that are connected to the system helium manifolds without overpressure protection devices in the connection path, from 2400 psig to 2200 psig, to match the design pressure of the helium manifolds - Increase the design temperature of a portion of the oxygen supply line L28 that directly interfaces with the recombination unit supply line L29, from 90°F to 250°F, to match the design temperature of the interfacing Class 2 portions of the Moderator Cover Gas system (L28 consists of the same 3/8 inch L101 tubing as L29). - Lower the design pressure of line L33, from the outlet of PRV49 and downstream, that interface with the PHT Storage, Transfer and Recovery system, from 2400 psig to 50 psig, to match the design pressure of the PHT Storage, Transfer and Recovery system (technical justification based on NK21-CALC-32310-00007).
Applicable Units	Unit 3
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Class 6 Moderator Cover Gas Overpressure Protection Modifications will have an immediate impact on maintaining the design basis of Class 6

	Moderator Cover Gas Overpressure Protection.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-100
GIO Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
CARD(s) Associated with this GIO	CA-0354, CA-0355, CA-0356, CA-0357, CA-0358, CA-0359
Gap(s) Associated with this CARD	SF02-SUP-024-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0355
CARD Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 4
CARD Description	The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 4 includes the following physical modifications: - Replacement of pressure regulating valve PRV28 - Installation of a new relief valve RV80 on L27, set at 220 psig, downstream of PRV28 - Installation of a new pressure regulating valve PRV59 on line L34, upstream of PRV44 (simply used to limit the maximum flow through PRV44 in case it ever fails open) - Replacement of RV45 on line L34 with a model that is ASME code-compliant - Installation of a new relief valve RV511 on line L33, set at 50 psig, downstream of PRV49 (to protect the interfacing PHTS D2O Storage, Transfer and Recovery system from overpressure) The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 4 includes the following non-physical modifications: - Increase the design pressure of various Class 6 portions of the Moderator Cover Gas system that directly interface with the Bulk Helium Supply system, from 150 psig to 220 psig, to match the design pressure of the Bulk Helium Supply system (technical justification based on NK21-CALC-32310-00007 and NK21-CALC-32310-00008. - Lower the design pressure of various Class 6 portions of the Moderator Cover Gas system that are connected to the system helium manifolds without overpressure protection devices in the connection path, from 2400 psig to 2200 psig, to match the design pressure of the helium manifolds - Increase the design temperature of a portion of the oxygen supply line L28 that directly interfaces with the recombination unit supply line L29, from 90°F to 250°F, to match the design temperature of the interfacing Class 2 portions of the Moderator Cover Gas system (L28 consists of the same 3/8 inch L101 tubing as L29). - Lower the design pressure of line L33, from the outlet of PRV49 and downstream, that interface with the PHT Storage, Transfer and Recovery system, from 2400 psig to 50 psig, to match the design pressure of the PHT Storage, Transfer and Recovery system (technical justification based on NK21-CALC-32310-00007).
Applicable Units	Unit 4
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Class 6 Moderator Cover Gas Overpressure Protection Modifications will have an immediate impact on maintaining the design basis of Class 6

	Moderator Cover Gas Overpressure Protection.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-100
GIO Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
CARD(s) Associated with this GIO	CA-0354, CA-0355, CA-0356, CA-0357, CA-0358, CA-0359
Gap(s) Associated with this CARD	SF02-SUP-025-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0356
CARD Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 5
CARD Description	The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 5 includes the following physical modifications: - Replacement of pressure regulating valve PRV28 - Installation of a new relief valve RV80 on L35, set at 220 psig, downstream of PRV28 - Installation of a new relief valve RV58 on L24, set at 15 psig, downstream of PRV36 - Installation of a new pressure regulating valve PRV59 on the bulk helium supply line, upstream of PRV41 (simply used to limit the maximum flow through PRV41 in case it ever fails open) - Installation of RV45 on line L34, set at 15 psig, downstream of PRV41
Applicable Units	Unit 5
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Class 6 Moderator Cover Gas Overpressure Protection Modifications will have an immediate impact on maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-100
GIO Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
CARD(s) Associated with this GIO	CA-0354, CA-0355, CA-0356, CA-0357, CA-0358, CA-0359
Gap(s) Associated with this CARD	SF02-SUP-026-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.

References	

CARD #	CA-0357
CARD Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 6
CARD Description	The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 6 includes the following physical modifications: - Replacement of pressure regulating valve PRV28 - Installation of a new relief valve RV80 on L35, set at 220 psig, downstream of PRV28 - Installation of a new relief valve RV58 on L24, set at 15 psig, downstream of PRV36 - Installation of a new pressure regulating valve PRV59 on the bulk helium supply line, upstream of PRV41 (simply used to limit the maximum flow through PRV41 in case it ever fails open) - Installation of RV45 on line L34, set at 15 psig, downstream of PRV41
Applicable Units	Unit 6
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Class 6 Moderator Cover Gas Overpressure Protection Modifications will have an immediate impact on maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-100
GIO Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
CARD(s) Associated with this GIO	CA-0354, CA-0355, CA-0356, CA-0357, CA-0358, CA-0359
Gap(s) Associated with this CARD	SF02-SUP-027-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.

References	

CARD #	CA-0358
CARD Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 7
CARD Description	The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 7 includes the following physical modifications: - Replacement of pressure regulating valve PRV28 - Installation of a new relief valve RV80 on L35, set at 220 psig, downstream of PRV28 - Installation of a new relief valve RV58 on L24, set at 15 psig, downstream of PRV36 - Installation of a new pressure regulating valve PRV59 on the bulk helium supply line, upstream of PRV41 (simply used to limit the maximum flow through PRV41 in case it ever fails open) - Installation of RV45 on line L34, set at 15 psig, downstream of PRV41
Applicable Units	Unit 7
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Class 6 Moderator Cover Gas Overpressure Protection Modifications will have an immediate impact on maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection . Column 2- Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-100
GIO Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
CARD(s) Associated with this GIO	CA-0354, CA-0355, CA-0356, CA-0357, CA-0358, CA-0359
Gap(s) Associated with this CARD	SF02-SUP-028-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.

References	

CARD #	CA-0359
CARD Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications - Unit 8
CARD Description	The Class 6 modifications to the Moderator Cover Gas piping circuit in Unit 8 includes the following physical modifications: - Replacement of pressure regulating valve PRV28 - Installation of a new relief valve RV80 on L35, set at 220 psig, downstream of PRV28 - Installation of a new relief valve RV58 on L24, set at 15 psig, downstream of PRV36 - Installation of a new pressure regulating valve PRV59 on the bulk helium supply line, upstream of PRV41 (simply used to limit the maximum flow through PRV41 in case it ever fails open) - Installation of RV45 on line L34, set at 15 psig, downstream of PRV41
Applicable Units	Unit 8
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Class 6 Moderator Cover Gas Overpressure Protection Modifications will have an immediate impact on maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Class 6 Moderator Cover Gas Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-100
GIO Title	M/32310 Class 6 Moderator Cover Gas Overpressure Protection Modifications
CARD(s) Associated with this GIO	CA-0354, CA-0355, CA-0356, CA-0357, CA-0358, CA-0359
Gap(s) Associated with this CARD	SF02-SUP-029-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.

References	

CARD #	CA-0360
CARD Title	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 1
CARD Description	The scope of the Design Change Notice, as per KDF (45784-009:070414), is to install 2 redundant relief valves on line L9D8, inside containment between the containment wall and 34720-MV5. The valves are to be provided with 3-way ball valve to facilitate isolation for maintenance and the discharge from the relief valves are routed to the moderator pit inside containment. Means to remotely confirm relief valve position (open/close),to comply with the 2007 edition of NB-7000, by preferably procuring relief valve complete with the limit switches for open and closed position and routing the signal to the MCR annunciation system.
Applicable Units	Unit 1
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-030-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0361
CARD Title	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 2
CARD Description	The scope of the Design Change Notice, as per KDF (45784-009:070414), is to install 2 redundant relief valves on line L9D8, inside containment between the containment wall and 34720-MV5. The valves are to be provided with 3-way ball valve to facilitate isolation for maintenance and the discharge from the relief valves are routed to the moderator pit inside containment. Means to remotely confirm relief valve position (open/close),to comply with the 2007 edition of NB-7000, by preferably procuring relief valve complete with the limit switches for open and closed position and routing the signal to the MCR annunciation system.
Applicable Units	Unit 2
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-031-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0362
CARD Title	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 3
CARD Description	The scope of the Design Change Notice, as per KDF (45784-009:070414), is to install 2 redundant relief valves on line L9D8, inside containment between the containment wall and 34720-MV5. The valves are to be provided with 3-way ball valve to facilitate isolation for maintenance and the discharge from the relief valves are routed to the moderator pit inside containment. Means to remotely confirm relief valve position (open/close),to comply with the 2007 edition of NB-7000, by preferably procuring relief valve complete with the limit switches for open and closed position and routing the signal to the MCR annunciation system.
Applicable Units	Unit 3
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-032-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0363
CARD Title	M/34720 Addition of Relief Valves For Overpressure Protection - Unit 4
CARD Description	The scope of the Design Change Notice, as per KDF (45784-009:070414), is to install 2 redundant relief valves on line L9D8, inside containment between the containment wall and 34720-MV5. The valves are to be provided with 3-way ball valve to facilitate isolation for maintenance and the discharge from the relief valves are routed to the moderator pit inside containment. Means to remotely confirm relief valve position (open/close),to comply with the 2007 edition of NB-7000, by preferably procuring relief valve complete with the limit switches for open and closed position and routing the signal to the MCR annunciation system.
Applicable Units	Unit 4
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-033-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0364
CARD Title	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 5
CARD Description	The scope of the modification under DCP 70276 is to modify the Maintenance Cooling System in each unit at Bruce B to mitigate the risk of unstable behavior as well as obtaining any code variances from the CNSC or correcting any non-compliance with ASME III, NB-7000.
Applicable Units	Unit 5
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-034-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0365
CARD Title	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 6
CARD Description	The scope of the modification under DCP 70276 is to modify the Maintenance Cooling System in each unit at Bruce B to mitigate the risk of unstable behavior as well as obtaining any code variances from the CNSC or correcting any non-compliance with ASME III, NB-7000.
Applicable Units	Unit 6
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system’s design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-035-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0366
CARD Title	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 7
CARD Description	The scope of the modification under DCP 70276 is to modify the Maintenance Cooling System in each unit at Bruce B to mitigate the risk of unstable behavior as well as obtaining any code variances from the CNSC or correcting any non-compliance with ASME III, NB-7000.
Applicable Units	Unit 7
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-036-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0367
CARD Title	M/34720 Replacement of Relief Valves For Overpressure Protection - Unit 8
CARD Description	The scope of the modification under DCP 70276 is to modify the Maintenance Cooling System in each unit at Bruce B to mitigate the risk of unstable behavior as well as obtaining any code variances from the CNSC or correcting any non-compliance with ASME III, NB-7000.
Applicable Units	Unit 8
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of Maintenance Cooling System Relief Valves For Overpressure Protection will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-101
GIO Title	M/34720 Relief Valves For Overpressure Protection
CARD(s) Associated with this GIO	CA-0360, CA-0361, CA-0362, CA-0363, CA-0364, CA-0365, CA-0366, CA-0367
Gap(s) Associated with this CARD	SF02-SUP-037-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0368
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 1
CARD Description	Modifications to achieve the monitoring of the opening and closing of the additional relief valves is as follows: Each RV will contain two limit switches. A limit switch for the fully open position shall be provided and will be used to monitor when the valve is fully open in accordance to the ASME requirement. A limit switch for the fully closed position shall be provided and will be used to monitor the valve close position in accordance to the ASME requirement. The switch logic will be arranged to annunciate: • RELIEF VALVE FULLY OPEN when the valve is fully open • RELIEF VALVE NOT FULLY CLOSE when the valve is not fully close When the valve is cracked open, the RELIEF VALVE NOT FULLY CLOSE annunciation will be activated. When the valve is fully open, both the RELIEF VALVE FULLY OPEN and the RELIEF VALVE NOT FULLY CLOSE annunciations will be activated. When the valve is fully close, all annunciations will be reset. A one minute timer will be used with the close limit switch signal to filter signals caused by valve chattering. Limit switches 34720-RV1-NS1 and 34720-RV1-NS2 will be wired/cabled from 34720-RV1 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB553 (about 100 ft). OLW will provide the spare wires to use up to the CDF. Limit switches 34720-RV2-NS1 and 34720-RV2-NS2 will be cabled from 34720-RV2 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB560 (about 100 ft). OLW will provide the spare wires to use up to the CDF.
Applicable Units	Unit 1
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000

Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-038-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0369
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 2
CARD Description	Modifications to achieve the monitoring of the opening and closing of the additional relief valves is as follows: Each RV will contain two limit switches. A limit switch for the fully open position shall be provided and will be used to monitor when the valve is fully open in accordance to the ASME requirement. A limit switch for the fully closed position shall be provided and will be used to monitor the valve close position in accordance to the ASME requirement. The switch logic will be arranged to annunciate: • RELIEF VALVE FULLY OPEN when the valve is fully open • RELIEF VALVE NOT FULLY CLOSE when the valve is not fully close When the valve is cracked open, the RELIEF VALVE NOT FULLY CLOSE annunciation will be activated. When the valve is fully open, both the RELIEF VALVE FULLY OPEN and the RELIEF VALVE NOT FULLY CLOSE annunciations will be activated. When the valve is fully close, all annunciations will be reset. A one minute timer will be used with the close limit switch signal to filter signals caused by valve chattering. Limit switches 34720-RV1-NS1 and 34720-RV1-NS2 will be wired/cabled from 34720-RV1 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB553 (about 100 ft). OLW will provide the spare wires to use up to the CDF. Limit switches 34720-RV2-NS1 and 34720-RV2-NS2 will be cabled from 34720-RV2 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB560 (about 100 ft). OLW will provide the spare wires to use up to the CDF.
Applicable Units	Unit 2
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000

Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-039-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0370
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 3
CARD Description	Modifications to achieve the monitoring of the opening and closing of the additional relief valves is as follows: Each RV will contain two limit switches. A limit switch for the fully open position shall be provided and will be used to monitor when the valve is fully open in accordance to the ASME requirement. A limit switch for the fully closed position shall be provided and will be used to monitor the valve close position in accordance to the ASME requirement. The switch logic will be arranged to annunciate: • RELIEF VALVE FULLY OPEN when the valve is fully open • RELIEF VALVE NOT FULLY CLOSE when the valve is not fully close When the valve is cracked open, the RELIEF VALVE NOT FULLY CLOSE annunciation will be activated. When the valve is fully open, both the RELIEF VALVE FULLY OPEN and the RELIEF VALVE NOT FULLY CLOSE annunciations will be activated. When the valve is fully close, all annunciations will be reset. A one minute timer will be used with the close limit switch signal to filter signals caused by valve chattering. Limit switches 34720-RV1-NS1 and 34720-RV1-NS2 will be wired/cabled from 34720-RV1 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB553 (about 100 ft). OLW will provide the spare wires to use up to the CDF. Limit switches 34720-RV2-NS1 and 34720-RV2-NS2 will be cabled from 34720-RV2 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB560 (about 100 ft). OLW will provide the spare wires to use up to the CDF.
Applicable Units	Unit 3
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system’s design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000

Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-040-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0371
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 4
CARD Description	Modifications to achieve the monitoring of the opening and closing of the additional relief valves is as follows: Each RV will contain two limit switches. A limit switch for the fully open position shall be provided and will be used to monitor when the valve is fully open in accordance to the ASME requirement. A limit switch for the fully closed position shall be provided and will be used to monitor the valve close position in accordance to the ASME requirement. The switch logic will be arranged to annunciate: • RELIEF VALVE FULLY OPEN when the valve is fully open • RELIEF VALVE NOT FULLY CLOSE when the valve is not fully close When the valve is cracked open, the RELIEF VALVE NOT FULLY CLOSE annunciation will be activated. When the valve is fully open, both the RELIEF VALVE FULLY OPEN and the RELIEF VALVE NOT FULLY CLOSE annunciations will be activated. When the valve is fully close, all annunciations will be reset. A one minute timer will be used with the close limit switch signal to filter signals caused by valve chattering. Limit switches 34720-RV1-NS1 and 34720-RV1-NS2 will be wired/cabled from 34720-RV1 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB553 (about 100 ft). OLW will provide the spare wires to use up to the CDF. Limit switches 34720-RV2-NS1 and 34720-RV2-NS2 will be cabled from 34720-RV2 up to a local junction box with Quick disconnect pigtails in between and then cabled to existing JB560 (about 100 ft). OLW will provide the spare wires to use up to the CDF.
Applicable Units	Unit 4
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000

Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-041-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0372
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 5
CARD Description	In accordance with ASME Section III - 1974, there currently is no instrumentation provided on the existing relief valves that provides indication of valve position. Upon modification of the RVs there will be a requirement of ASME Section III - 2010 NB-7131 (b) as follows: "Means shall be provided for remote monitoring of valve position (fully open and fully closed). These means may be incorporated in the valve design or its system installation." The scope of this CARD is to provide instrumentation of the relief valves upon the RVs being modified to meet the new requirement. Alternatively; Bruce Power may seek formal CNSC approval for a code concession.
Applicable Units	Unit 5
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-042-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0373
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 6
CARD Description	In accordance with ASME Section III - 1974, there currently is no instrumentation provided on the existing relief valves that provides indication of valve position. Upon modification of the RVs there will be a requirement of ASME Section III - 2010 NB-7131 (b) as follows: "Means shall be provided for remote monitoring of valve position (fully open and fully closed). These means may be incorporated in the valve design or its system installation." The scope of this CARD is to provide instrumentation of the relief valves upon the RVs being modified to meet the new requirement. Alternatively; Bruce Power may seek formal CNSC approval for a code concession.
Applicable Units	Unit 6
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-043-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0374
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 7
CARD Description	In accordance with ASME Section III - 1974, there currently is no instrumentation provided on the existing relief valves that provides indication of valve position. Upon modification of the RVs there will be a requirement of ASME Section III - 2010 NB-7131 (b) as follows: "Means shall be provided for remote monitoring of valve position (fully open and fully closed). These means may be incorporated in the valve design or its system installation." The scope of this CARD is to provide instrumentation of the relief valves upon the RVs being modified to meet the new requirement. Alternatively; Bruce Power may seek formal CNSC approval for a code concession.
Applicable Units	Unit 7
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system's design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-044-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0375
CARD Title	I/63472 Remote Relief Valve Position Indication - Unit 8
CARD Description	In accordance with ASME Section III - 1974, there currently is no instrumentation provided on the existing relief valves that provides indication of valve position. Upon modification of the RVs there will be a requirement of ASME Section III - 2010 NB-7131 (b) as follows: "Means shall be provided for remote monitoring of valve position (fully open and fully closed). These means may be incorporated in the valve design or its system installation." The scope of this CARD is to provide instrumentation of the relief valves upon the RVs being modified to meet the new requirement. Alternatively; Bruce Power may seek formal CNSC approval for a code concession.
Applicable Units	Unit 8
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Addition of Maintenance Cooling System Remote Relief Valve Position Indication will have an immediate impact on maintaining the system’s design basis.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of Maintenance Cooling System Overpressure Protection. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.49160
CARD Priority	1
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-102
GIO Title	I/63472 Remote Relief Valve Position Indication
CARD(s) Associated with this GIO	CA-0368, CA-0369, CA-0370, CA-0371, CA-0372, CA-0373, CA-0374, CA-0375
Gap(s) Associated with this CARD	SF02-SUP-045-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note this is a milestone date, upon completion of milestone TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0084
CARD Title	In-Service Inspection Program for Bruce NGS A and B Safety Related Structures
CARD Description	Update NK21-PIP-20000-00001 and NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Safety Related Structure Bruce NGS A and B to describe inspection requirements following an abnormal/environmental condition in accordance with Clause 7.3.4 of CSA N291-15.
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMES (SECCVD)
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Resolution of the issue(s) through completion of this CA will take up to 3 years to have its effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix- Row 2, Column 2 Inclusion of inspection requirements following an abnormal/environmental condition will augment operational safety and performance of safety related structures.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-039
GIO Title	Equipment Reliability and Maintenance
CARD(s) Associated with this GIO	CA-0084
Gap(s) Associated with this CARD	SF04_CSA 291-15_7.3.4_16, SF04_CSA N291-08_7.3.4_15
Additional Information	This will be implemented by PSE-OMA-81136 project.
References	

CARD #	CA-0123
CARD Title	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 6
CARD Description	Scope of work covers inspections and analysis of the major structural elements of the Calandria and Shield Tank Assembly (CSTA) to identify extent of (if any) ageing degradation due to leakage, radiation embrittlement, corrosion, mechanical damage and foreign material. The following areas will be inspected, cleaned and repaired as necessary: - Calandria Main Shell - Calandria Subshell - Annular Plate - Calandria Tubesheet - Lattice Tube - Calandria Internal Surfaces - Moderator Inlet Nozzles - Reactivity Control Unit Guide Tubes - Liquid Injection Shutdown System (LISS) Nozzles - Absorber Elements - Liquid Zone Control Unit (LZCU) - Calandria Pressure Relief Duct Piping - Reactivity Mechanisms - Moderator Piping
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Calandra and Shield Tank Assembly (CSTA) Major Inspection will have an immediate impact on understanding the condition of CSTA.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of CSTA. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-059
GIO Title	Calandria and Shield Tank Assembly Major Inspection
CARD(s) Associated with this GIO	CA-0123, CA-0212, CA-0229, CA-0246, CA-0263, CA-0280

Gap(s) Associated with this CARD	SF04-MCR-0004-16
Additional Information	AMOT-0208
References	

CARD #	CA-0212
CARD Title	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 3
CARD Description	Scope of work covers inspections and analysis of the major structural elements of the Calandria and Shield Tank Assembly (CSTA) to identify extent of (if any) ageing degradation due to leakage, radiation embrittlement, corrosion, mechanical damage and foreign material. The following areas will be inspected, cleaned and repaired as necessary: - Calandria Main Shell - Calandria Subshell - Annular Plate - Calandria Tubesheet - Lattice Tube - Calandria Internal Surfaces - Moderator Inlet Nozzles - Reactivity Control Unit Guide Tubes - Liquid Injection Shutdown System (LISS) Nozzles - Absorber Elements - Liquid Zone Control Unit (LZCU) - Calandria Pressure Relief Duct Piping - Reactivity Mechanisms - Moderator Piping
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Calandra and Shield Tank Assembly (CSTA) Major Inspection will have an immediate impact on understanding the condition of CSTA.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of CSTA. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-059
GIO Title	Calandria and Shield Tank Assembly Major Inspection
CARD(s) Associated with this GIO	CA-0123, CA-0212, CA-0229, CA-0246, CA-0263, CA-0280

Gap(s) Associated with this CARD	SF02-MCR-0035-16
Additional Information	AMOT-0207
References	

CARD #	CA-0229
CARD Title	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 4
CARD Description	Scope of work covers inspections and analysis of the major structural elements of the Calandria and Shield Tank Assembly (CSTA) to identify extent of (if any) ageing degradation due to leakage, radiation embrittlement, corrosion, mechanical damage and foreign material. The following areas will be inspected, cleaned and repaired as necessary: - Calandria Main Shell - Calandria Subshell - Annular Plate - Calandria Tubesheet - Lattice Tube - Calandria Internal Surfaces - Moderator Inlet Nozzles - Reactivity Control Unit Guide Tubes - Liquid Injection Shutdown System (LISS) Nozzles - Absorber Elements - Liquid Zone Control Unit (LZCU) - Calandria Pressure Relief Duct Piping - Reactivity Mechanisms - Moderator Piping
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Calandra and Shield Tank Assembly (CSTA) Major Inspection will have an immediate impact on understanding the condition of CSTA.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of CSTA. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-059
GIO Title	Calandria and Shield Tank Assembly Major Inspection
CARD(s) Associated with this GIO	CA-0123, CA-0212, CA-0229, CA-0246, CA-0263, CA-0280

Gap(s) Associated with this CARD	SF02-MCR-0052-16
Additional Information	AMOT-0207
References	

CARD #	CA-0246
CARD Title	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 5
CARD Description	Scope of work covers inspections and analysis of the major structural elements of the Calandria and Shield Tank Assembly (CSTA) to identify extent of (if any) ageing degradation due to leakage, radiation embrittlement, corrosion, mechanical damage and foreign material. The following areas will be inspected, cleaned and repaired as necessary: - Calandria Main Shell - Calandria Subshell - Annular Plate - Calandria Tubesheet - Lattice Tube - Calandria Internal Surfaces - Moderator Inlet Nozzles - Reactivity Control Unit Guide Tubes - Liquid Injection Shutdown System (LISS) Nozzles - Absorber Elements - Liquid Zone Control Unit (LZCU) - Calandria Pressure Relief Duct Piping - Reactivity Mechanisms - Moderator Piping
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Calandra and Shield Tank Assembly (CSTA) Major Inspection will have an immediate impact on understanding the condition of CSTA.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of CSTA. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-059
GIO Title	Calandria and Shield Tank Assembly Major Inspection
CARD(s) Associated with this GIO	CA-0123, CA-0212, CA-0229, CA-0246, CA-0263, CA-0280

Gap(s) Associated with this CARD	SF02-MCR-0069-16
Additional Information	AMOT-0208
References	

CARD #	CA-0263
CARD Title	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 7
CARD Description	Scope of work covers inspections and analysis of the major structural elements of the Calandria and Shield Tank Assembly (CSTA) to identify extent of (if any) ageing degradation due to leakage, radiation embrittlement, corrosion, mechanical damage and foreign material. The following areas will be inspected, cleaned and repaired as necessary: - Calandria Main Shell - Calandria Subshell - Annular Plate - Calandria Tubesheet - Lattice Tube - Calandria Internal Surfaces - Moderator Inlet Nozzles - Reactivity Control Unit Guide Tubes - Liquid Injection Shutdown System (LISS) Nozzles - Absorber Elements - Liquid Zone Control Unit (LZCU) - Calandria Pressure Relief Duct Piping - Reactivity Mechanisms - Moderator Piping
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Calandra and Shield Tank Assembly (CSTA) Major Inspection will have an immediate impact on understanding the condition of CSTA.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of CSTA. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-059
GIO Title	Calandria and Shield Tank Assembly Major Inspection
CARD(s) Associated with this GIO	CA-0123, CA-0212, CA-0229, CA-0246, CA-0263, CA-0280

Gap(s) Associated with this CARD	SF02-MCR-0086-16
Additional Information	AMOT-0208
References	

CARD #	CA-0280
CARD Title	Calandria and Shield Tank Assembly (CSTA) Major Inspection - Unit 8
CARD Description	Scope of work covers inspections and analysis of the major structural elements of the Calandria and Shield Tank Assembly (CSTA) to identify extent of (if any) ageing degradation due to leakage, radiation embrittlement, corrosion, mechanical damage and foreign material. The following areas will be inspected, cleaned and repaired as necessary: - Calandria Main Shell - Calandria Subshell - Annular Plate - Calandria Tubesheet - Lattice Tube - Calandria Internal Surfaces - Moderator Inlet Nozzles - Reactivity Control Unit Guide Tubes - Liquid Injection Shutdown System (LISS) Nozzles - Absorber Elements - Liquid Zone Control Unit (LZCU) - Calandria Pressure Relief Duct Piping - Reactivity Mechanisms - Moderator Piping
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Calandra and Shield Tank Assembly (CSTA) Major Inspection will have an immediate impact on understanding the condition of CSTA.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of CSTA. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-059
GIO Title	Calandria and Shield Tank Assembly Major Inspection
CARD(s) Associated with this GIO	CA-0123, CA-0212, CA-0229, CA-0246, CA-0263, CA-0280

Gap(s) Associated with this CARD	SF02-MCR-0103-16
Additional Information	AMOT-0208
References	

CARD #	CA-0124
CARD Title	Preheater Inspections - Unit 6
CARD Description	The scope of inspection and maintenance activities include: - Eddy Current inspection of 2 PHs - Ultrasonic Testing and ADAM probe inspection of a sample of tubes in 1 PH - Removal and reinstallation of drain lines for all 4 PHs to facilitate visual inspection of u-bend support assemblies - Removal of two tubes from 1 PH for metallographic analysis and visual inspection of the secondary tubesheet face and supports following tube removals. - Severing and rejoining of all 4 PHs to facilitate secondary side visual inspection of peripheral tubes, feedwater inlet and outlet assemblies and tube supports. - Resurfacing of all gasket sealing surfaces - Inspection of all manway cover studs and nuts - Primary head visual inspections (locking tabs, divider plates, tubesheet surfaces, tubesheet plugs and divider plate ear pieces - Planned divider plate replacement and/or sealing skin installation in all 4 PHs - Tube plugging as required
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Preheater Inspections will have an immediate impact on understanding the condition of the vessels.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Preheater. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596

CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-060
GIO Title	Preheater Inspections
CARD(s) Associated with this GIO	CA-0124, CA-0213, CA-0230, CA-0247, CA-0264, CA-0281
Gap(s) Associated with this CARD	SF02-MCR-0005-16
Additional Information	AMOT-0011
References	

CARD #	CA-0213
CARD Title	Preheater Inspections - Unit 3
CARD Description	The scope of inspection and maintenance activities include: - Eddy Current inspection of 2 PHs - Ultrasonic Testing and ADAM probe inspection of a sample of tubes in 1 PH - Removal and reinstallation of drain lines for all 4 PHs to facilitate visual inspection of u-bend support assemblies - Removal of two tubes from 1 PH for metallographic analysis and visual inspection of the secondary tubesheet face and supports following tube removals. - Severing and rejoining of all 4 PHs to facilitate secondary side visual inspection of peripheral tubes, feedwater inlet and outlet assemblies and tube supports. - Resurfacing of all gasket sealing surfaces - Inspection of all manway cover studs and nuts - Primary head visual inspections (locking tabs, divider plates, tubesheet surfaces, tubesheet plugs and divider plate ear pieces - Planned divider plate replacement and/or sealing skin installation in all 4 PHs - Tube plugging as required
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Preheater Inspections will have an immediate impact on understanding the condition of the vessels.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Preheater. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596

CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-060
GIO Title	Preheater Inspections
CARD(s) Associated with this GIO	CA-0124, CA-0213, CA-0230, CA-0247, CA-0264, CA-0281
Gap(s) Associated with this CARD	SF02-MCR-0036-16
Additional Information	AMOT-0008
References	

CARD #	CA-0230
CARD Title	Preheater Inspections - Unit 4
CARD Description	The scope of inspection and maintenance activities include: - Eddy Current inspection of 2 PHs - Ultrasonic Testing and ADAM probe inspection of a sample of tubes in 1 PH - Removal and reinstallation of drain lines for all 4 PHs to facilitate visual inspection of u-bend support assemblies - Removal of two tubes from 1 PH for metallographic analysis and visual inspection of the secondary tubesheet face and supports following tube removals. - Severing and rejoining of all 4 PHs to facilitate secondary side visual inspection of peripheral tubes, feedwater inlet and outlet assemblies and tube supports. - Resurfacing of all gasket sealing surfaces - Inspection of all manway cover studs and nuts - Primary head visual inspections (locking tabs, divider plates, tubesheet surfaces, tubesheet plugs and divider plate ear pieces - Planned divider plate replacement and/or sealing skin installation in all 4 PHs - Tube plugging as required
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Preheater Inspections will have an immediate impact on understanding the condition of the vessels.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Preheater. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596

CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-060
GIO Title	Preheater Inspections
CARD(s) Associated with this GIO	CA-0124, CA-0213, CA-0230, CA-0247, CA-0264, CA-0281
Gap(s) Associated with this CARD	SF02-MCR-0053-16
Additional Information	AMOT-0008
References	

CARD #	CA-0247
CARD Title	Preheater Inspections - Unit 5
CARD Description	The scope of inspection and maintenance activities include: - Eddy Current inspection of 2 PHs - Ultrasonic Testing and ADAM probe inspection of a sample of tubes in 1 PH - Removal and reinstallation of drain lines for all 4 PHs to facilitate visual inspection of u-bend support assemblies - Removal of two tubes from 1 PH for metallographic analysis and visual inspection of the secondary tubesheet face and supports following tube removals. - Severing and rejoining of all 4 PHs to facilitate secondary side visual inspection of peripheral tubes, feedwater inlet and outlet assemblies and tube supports. - Resurfacing of all gasket sealing surfaces - Inspection of all manway cover studs and nuts - Primary head visual inspections (locking tabs, divider plates, tubesheet surfaces, tubesheet plugs and divider plate ear pieces - Planned divider plate replacement and/or sealing skin installation in all 4 PHs - Tube plugging as required
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Preheater Inspections will have an immediate impact on understanding the condition of the vessels.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Preheater. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596

CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-060
GIO Title	Preheater Inspections
CARD(s) Associated with this GIO	CA-0124, CA-0213, CA-0230, CA-0247, CA-0264, CA-0281
Gap(s) Associated with this CARD	SF02-MCR-0070-16
Additional Information	AMOT-0011
References	

CARD #	CA-0264
CARD Title	Preheater Inspections - Unit 7
CARD Description	The scope of inspection and maintenance activities include: - Eddy Current inspection of 2 PHs - Ultrasonic Testing and ADAM probe inspection of a sample of tubes in 1 PH - Removal and reinstallation of drain lines for all 4 PHs to facilitate visual inspection of u-bend support assemblies - Removal of two tubes from 1 PH for metallographic analysis and visual inspection of the secondary tubesheet face and supports following tube removals. - Severing and rejoining of all 4 PHs to facilitate secondary side visual inspection of peripheral tubes, feedwater inlet and outlet assemblies and tube supports. - Resurfacing of all gasket sealing surfaces - Inspection of all manway cover studs and nuts - Primary head visual inspections (locking tabs, divider plates, tubesheet surfaces, tubesheet plugs and divider plate ear pieces - Planned divider plate replacement and/or sealing skin installation in all 4 PHs - Tube plugging as required
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Preheater Inspections will have an immediate impact on understanding the condition of the vessels.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Preheater. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596

CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-060
GIO Title	Preheater Inspections
CARD(s) Associated with this GIO	CA-0124, CA-0213, CA-0230, CA-0247, CA-0264, CA-0281
Gap(s) Associated with this CARD	SF02-MCR-0087-16
Additional Information	AMOT-0011
References	

CARD #	CA-0281
CARD Title	Preheater Inspections - Unit 8
CARD Description	The scope of inspection and maintenance activities include: - Eddy Current inspection of 2 PHs - Ultrasonic Testing and ADAM probe inspection of a sample of tubes in 1 PH - Removal and reinstallation of drain lines for all 4 PHs to facilitate visual inspection of u-bend support assemblies - Removal of two tubes from 1 PH for metallographic analysis and visual inspection of the secondary tubesheet face and supports following tube removals. - Severing and rejoining of all 4 PHs to facilitate secondary side visual inspection of peripheral tubes, feedwater inlet and outlet assemblies and tube supports. - Resurfacing of all gasket sealing surfaces - Inspection of all manway cover studs and nuts - Primary head visual inspections (locking tabs, divider plates, tubesheet surfaces, tubesheet plugs and divider plate ear pieces - Planned divider plate replacement and/or sealing skin installation in all 4 PHs - Tube plugging as required
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Preheater Inspections will have an immediate impact on understanding the condition of the vessels.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Preheater. Column 2-Augments/ recovers the current understanding of the physical condition.
Time-Impact Utility Score	0.70374
Final Score	0.34596

CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-060
GIO Title	Preheater Inspections
CARD(s) Associated with this GIO	CA-0124, CA-0213, CA-0230, CA-0247, CA-0264, CA-0281
Gap(s) Associated with this CARD	SF02-MCR-0104-16
Additional Information	AMOT-0011
References	

CARD #	CA-0129
CARD Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 6
CARD Description	The scope of work covers inspection, testing and necessary repairs/replacements of seismic restraints: - Visual inspection in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan - Functional drag testing of 10% of the snubbers visually inspected
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Seismic Restraint Inspections will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Seismic Restraints. Column 2-Augments/ recovers the current understanding of the physical condition and functionality of Seismic Restraints.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-065
GIO Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection
CARD(s) Associated with this GIO	CA-0129, CA-0336, CA-0337, CA-0338, CA-0339, CA-0340
Gap(s) Associated with this CARD	SF02-MCR-0010-16
Additional Information	U6-MCR-SOW-6092
References	

CARD #	CA-0336
CARD Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 3
CARD Description	The scope of work covers inspection, testing and necessary repairs/replacements of seismic restraints: - Visual inspection in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan - Functional drag testing of 10% of the snubbers visually inspected
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Seismic Restraint Inspections will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Seismic Restraints. Column 2-Augments/ recovers the current understanding of the physical condition and functionality of Seismic Restraints.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-065
GIO Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection
CARD(s) Associated with this GIO	CA-0129, CA-0336, CA-0337, CA-0338, CA-0339, CA-0340
Gap(s) Associated with this CARD	SF02-MCR-0132-16
Additional Information	
References	

CARD #	CA-0337
CARD Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 4
CARD Description	The scope of work covers inspection, testing and necessary repairs/replacements of seismic restraints: - Visual inspection in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan - Functional drag testing of 10% of the snubbers visually inspected
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Seismic Restraint Inspections will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Seismic Restraints. Column 2-Augments/ recovers the current understanding of the physical condition and functionality of Seismic Restraints.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-065
GIO Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection
CARD(s) Associated with this GIO	CA-0129, CA-0336, CA-0337, CA-0338, CA-0339, CA-0340
Gap(s) Associated with this CARD	SF02-MCR-0133-16
Additional Information	
References	

CARD #	CA-0338
CARD Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 5
CARD Description	The scope of work covers inspection, testing and necessary repairs/replacements of seismic restraints: - Visual inspection in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan - Functional drag testing of 10% of the snubbers visually inspected
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Seismic Restraint Inspections will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Seismic Restraints. Column 2-Augments/ recovers the current understanding of the physical condition and functionality of Seismic Restraints.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-065
GIO Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection
CARD(s) Associated with this GIO	CA-0129, CA-0336, CA-0337, CA-0338, CA-0339, CA-0340
Gap(s) Associated with this CARD	SF02-MCR-0134-16
Additional Information	
References	

CARD #	CA-0339
CARD Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 7
CARD Description	The scope of work covers inspection, testing and necessary repairs/replacements of seismic restraints: - Visual inspection in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan - Functional drag testing of 10% of the snubbers visually inspected
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Seismic Restraint Inspections will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Seismic Restraints. Column 2-Augments/ recovers the current understanding of the physical condition and functionality of Seismic Restraints.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-065
GIO Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection
CARD(s) Associated with this GIO	CA-0129, CA-0336, CA-0337, CA-0338, CA-0339, CA-0340
Gap(s) Associated with this CARD	SF02-MCR-0135-16
Additional Information	
References	

CARD #	CA-0340
CARD Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection - Unit 8
CARD Description	The scope of work covers inspection, testing and necessary repairs/replacements of seismic restraints: - Visual inspection in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan - Functional drag testing of 10% of the snubbers visually inspected
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Seismic Restraint Inspections will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of Seismic Restraints. Column 2-Augments/ recovers the current understanding of the physical condition and functionality of Seismic Restraints.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-065
GIO Title	PHT Seismic Restraints (Snubbers)-Periodic Inspection Program (PIP)-Inspection
CARD(s) Associated with this GIO	CA-0129, CA-0336, CA-0337, CA-0338, CA-0339, CA-0340
Gap(s) Associated with this CARD	SF02-MCR-0136-16
Additional Information	
References	

CARD #	CA-0130
CARD Title	Pressurizer and Supports- Internal Inspection - Unit 6
CARD Description	Perform inspections in accordance with CSA-N285.4 as defined in unit 6 Periodic Inspection Program (PIP) Plan: - Visual and surface inspection of vessel internals and supports - Ultrasonic wall thickness measurements at the vessel waterline, outlet nozzle and other areas of potential degradation
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 6- Pressurizer and Supports- Internal Inspection will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of the Pressurizer and its supports. Column 2-Augments/ recovers the current understanding of the physical condition the Pressurizer and its supports.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-066
GIO Title	Pressurizer and Supports- Internal Inspection
CARD(s) Associated with this GIO	CA-0130, CA-0341, CA-0342, CA-0343, CA-0344, CA-0345
Gap(s) Associated with this CARD	SF02-MCR-0011-16
Additional Information	U6-MCR-SOW-6094-R00
References	

CARD #	CA-0341
CARD Title	Pressurizer and Supports- Internal Inspection - Unit 5
CARD Description	Perform inspections in accordance with CSA-N285.4 as defined in unit 5 Periodic Inspection Program (PIP) Plan: - Visual and surface inspection of vessel internals and supports - Ultrasonic wall thickness measurements at the vessel waterline, outlet nozzle and other areas of potential degradation
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Pressurizer and Supports- Internal Inspection will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of the Pressurizer and its supports. Column 2-Augments/ recovers the current understanding of the physical condition the Pressurizer and its supports.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-066
GIO Title	Pressurizer and Supports- Internal Inspection
CARD(s) Associated with this GIO	CA-0130, CA-0341, CA-0342, CA-0343, CA-0344, CA-0345
Gap(s) Associated with this CARD	SF02-MCR-0137-16
Additional Information	
References	

CARD #	CA-0342
CARD Title	Pressurizer and Supports- Internal Inspection - Unit 7
CARD Description	Perform inspections in accordance with CSA-N285.4 as defined in unit 5 Periodic Inspection Program (PIP) Plan: - Visual and surface inspection of vessel internals and supports - Ultrasonic wall thickness measurements at the vessel waterline, outlet nozzle and other areas of potential degradation
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 7- Pressurizer and Supports- Internal Inspection will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of the Pressurizer and its supports. Column 2-Augments/ recovers the current understanding of the physical condition the Pressurizer and its supports.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-066
GIO Title	Pressurizer and Supports- Internal Inspection
CARD(s) Associated with this GIO	CA-0130, CA-0341, CA-0342, CA-0343, CA-0344, CA-0345
Gap(s) Associated with this CARD	SF02-MCR-0138-16
Additional Information	
References	

CARD #	CA-0343
CARD Title	Pressurizer and Supports- Internal Inspection - Unit 8
CARD Description	Perform inspections in accordance with CSA-N285.4 as defined in unit 8 Periodic Inspection Program (PIP) Plan: - Visual and surface inspection of vessel internals and supports - Ultrasonic wall thickness measurements at the vessel waterline, outlet nozzle and other areas of potential degradation
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 8- Pressurizer and Supports- Internal Inspection will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of the Pressurizer and its supports. Column 2-Augments/ recovers the current understanding of the physical condition the Pressurizer and its supports.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-066
GIO Title	Pressurizer and Supports- Internal Inspection
CARD(s) Associated with this GIO	CA-0130, CA-0341, CA-0342, CA-0343, CA-0344, CA-0345
Gap(s) Associated with this CARD	SF02-MCR-0139-16
Additional Information	
References	

CARD #	CA-0344
CARD Title	Pressurizer and Supports- Internal Inspection - Unit 3
CARD Description	Perform inspections in accordance with CSA-N285.4 as defined in unit 3 Periodic Inspection Program (PIP) Plan: - Visual and surface inspection of vessel internals and supports - Ultrasonic wall thickness measurements at the vessel waterline, outlet nozzle and other areas of potential degradation
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 5- Pressurizer and Supports- Internal Inspection will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of the Pressurizer and its supports. Column 2-Augments/ recovers the current understanding of the physical condition the Pressurizer and its supports.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-066
GIO Title	Pressurizer and Supports- Internal Inspection
CARD(s) Associated with this GIO	CA-0130, CA-0341, CA-0342, CA-0343, CA-0344, CA-0345
Gap(s) Associated with this CARD	SF02-MCR-0159-16
Additional Information	
References	

CARD #	CA-0345
CARD Title	Pressurizer and Supports- Internal Inspection - Unit 4
CARD Description	Perform inspections in accordance with CSA-N285.4 as defined in unit 4 Periodic Inspection Program (PIP) Plan: - Visual and surface inspection of vessel internals and supports - Ultrasonic wall thickness measurements at the vessel waterline, outlet nozzle and other areas of potential degradation
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Pressurizer and Supports- Internal Inspection will have an immediate impact on understanding the condition and functionality of the Seismic Restraints.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved understanding of the condition of the Pressurizer and its supports. Column 2-Augments/ recovers the current understanding of the physical condition the Pressurizer and its supports.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-066
GIO Title	Pressurizer and Supports- Internal Inspection
CARD(s) Associated with this GIO	CA-0130, CA-0341, CA-0342, CA-0343, CA-0344, CA-0345
Gap(s) Associated with this CARD	SF02-MCR-0160-16
Additional Information	
References	

CARD #	CA-0207
CARD Title	PHT Valves-Refurbishment of 33120-MV23 - Unit 6
CARD Description	Refurbish PHT Valve 33120-MV23. Scope of work covers replacement and overhaul of valve components including valve packing, linkage, 4 bevel gearboxes and 1 valve gearbox.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Refurbishment of 33120-MV23 will have an immediate impact on maintaining the design basis of 33120-MV23.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 33120-MV23. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-086
GIO Title	PHT Valves-Refurbishment of 33120-MV23
CARD(s) Associated with this GIO	CA-0207, CA-0224, CA-0241, CA-0258, CA-0275, CA-0292
Gap(s) Associated with this CARD	SF02-MCR-0158-16
Additional Information	38843-MCR6-SoW-002
References	

CARD #	CA-0224
CARD Title	PHT Valves-Refurbishment of 33120-MV23 - Unit 3
CARD Description	Refurbish PHT Valve 33120-MV23. Scope of work covers replacement and overhaul of valve components including valve packing, linkage, 4 bevel gearboxes and 1 valve gearbox.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Refurbishment of 33120-MV23 will have an immediate impact on maintaining the design basis of 33120-MV23.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 33120-MV23. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-086
GIO Title	PHT Valves-Refurbishment of 33120-MV23
CARD(s) Associated with this GIO	CA-0207, CA-0224, CA-0241, CA-0258, CA-0275, CA-0292
Gap(s) Associated with this CARD	SF02-MCR-0047-16
Additional Information	
References	

CARD #	CA-0241
CARD Title	PHT Valves-Refurbishment of 33120-MV23 - Unit 4
CARD Description	Refurbish PHT Valve 33120-MV23. Scope of work covers replacement and overhaul of valve components including valve packing, linkage, 4 bevel gearboxes and 1 valve gearbox.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Refurbishment of 33120-MV23 will have an immediate impact on maintaining the design basis of 33120-MV23.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 33120-MV23. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-086
GIO Title	PHT Valves-Refurbishment of 33120-MV23
CARD(s) Associated with this GIO	CA-0207, CA-0224, CA-0241, CA-0258, CA-0275, CA-0292
Gap(s) Associated with this CARD	SF02-MCR-0064-16
Additional Information	
References	

CARD #	CA-0258
CARD Title	PHT Valves-Refurbishment of 33120-MV23 - Unit 5
CARD Description	Refurbish PHT Valve 33120-MV23. Scope of work covers replacement and overhaul of valve components including valve packing, linkage, 4 bevel gearboxes and 1 valve gearbox.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Refurbishment of 33120-MV23 will have an immediate impact on maintaining the design basis of 33120-MV23.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 33120-MV23. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-086
GIO Title	PHT Valves-Refurbishment of 33120-MV23
CARD(s) Associated with this GIO	CA-0207, CA-0224, CA-0241, CA-0258, CA-0275, CA-0292
Gap(s) Associated with this CARD	SF02-MCR-0081-16
Additional Information	
References	

CARD #	CA-0275
CARD Title	PHT Valves-Refurbishment of 33120-MV23 - Unit 7
CARD Description	Refurbish PHT Valve 33120-MV23. Scope of work covers replacement and overhaul of valve components including valve packing, linkage, 4 bevel gearboxes and 1 valve gearbox.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Refurbishment of 33120-MV23 will have an immediate impact on maintaining the design basis of 33120-MV23.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 33120-MV23. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-086
GIO Title	PHT Valves-Refurbishment of 33120-MV23
CARD(s) Associated with this GIO	CA-0207, CA-0224, CA-0241, CA-0258, CA-0275, CA-0292
Gap(s) Associated with this CARD	SF02-MCR-0098-16
Additional Information	
References	

CARD #	CA-0292
CARD Title	PHT Valves-Refurbishment of 33120-MV23 - Unit 8
CARD Description	Refurbish PHT Valve 33120-MV23. Scope of work covers replacement and overhaul of valve components including valve packing, linkage, 4 bevel gearboxes and 1 valve gearbox.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Refurbishment of 33120-MV23 will have an immediate impact on maintaining the design basis of 33120-MV23.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 33120-MV23. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-086
GIO Title	PHT Valves-Refurbishment of 33120-MV23
CARD(s) Associated with this GIO	CA-0207, CA-0224, CA-0241, CA-0258, CA-0275, CA-0292
Gap(s) Associated with this CARD	SF02-MCR-0115-16
Additional Information	
References	

CARD #	CA-0321
CARD Title	45VDC Power Supplies-Replacement - Unit 0A
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 0A
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0117-16
Additional Information	AMOT-0169

References	

CARD #	CA-0322
CARD Title	45VDC Power Supplies-Replacement - Unit 0B
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 0B
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0118-16
Additional Information	AMOT-0170

References	

CARD #	CA-0323
CARD Title	45VDC Power Supplies-Replacement - Unit 3
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0119-16
Additional Information	AMOT-0169

References	

CARD #	CA-0324
CARD Title	45VDC Power Supplies-Replacement - Unit 4
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0120-16
Additional Information	AMOT-0169

References	

CARD #	CA-0325
CARD Title	45VDC Power Supplies-Replacement - Unit 5
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 5
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-29
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0121-16
Additional Information	AMOT-0170

References	

CARD #	CA-0326
CARD Title	45VDC Power Supplies-Replacement - Unit 6
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 6
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	31-Dec-23
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0122-16
Additional Information	AMOT-0170

References	

CARD #	CA-0327
CARD Title	45VDC Power Supplies-Replacement - Unit 7
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 7
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-31
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0123-16
Additional Information	AMOT-0170

References	

CARD #	CA-0328
CARD Title	45VDC Power Supplies-Replacement - Unit 8
CARD Description	The scope of work includes replacement of 45 Vdc power supplies with same fit, form, functional equivalents. 1) The 45 Vdc, 1 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. 2) The 45 Vdc, 1 Amp EQ (Seismically Qualified to the appropriate Design Basis Earthquake (DBE) level) power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS2, ECI and Containment systems. 3) The 45 Vdc, 2 Amp power supplies are regulated and used for supplying electrical power to control and monitoring of various systems within SDS1, SDS2, ECI, RRS and other systems. The scope of work also includes the refurbishment of each panel and may require replacement of wiring harness.
Applicable Units	Unit 8
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Replacement of 45VDC Power Supplies will have an immediate impact on maintaining the design basis of 45VDC Power Supplies.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of 45VDC Power Supplies. Column 2-Augments/ recovers the current barriers in the design.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Not Started
Target Completion Date	30-Jun-33
GIO #	GIO-095
GIO Title	45VDC Power Supplies-Replacement
CARD(s) Associated with this GIO	CA-0321, CA-0322, CA-0323, CA-0324, CA-0325, CA-0326, CA-0327, CA-0328
Gap(s) Associated with this CARD	SF02-MCR-0124-16
Additional Information	AMOT-0170

References	

CARD #	CA-0379
CARD Title	Bruce B Heat Transport Vibration Project
CARD Description	Examination of the potential benefits of the implementation of design modifications to the primary heat transport system (PHT) to alleviate PHT vibration impacts. Design modifications being considered include, but are not limited to the resonator inlet shield plug (RISP) and the 7-vane impellor for the PHT pumps.
Applicable Units	Bruce B
Alert Group	DPTPROJB
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Results of Bruce B Heat Transport Vibration Project will have an immediate impact on maintaining the design basis for fuel and fuel channel integrity.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Resolving the issue will improve safety and operational performance of fuel and fuel channels, Column 2- Augments recovers operational performance of fuel and fuel channels.
Time-Impact Utility Score	0.70374
Final Score	0.34596
CARD Priority	2
CARD Status	Underway
Target Completion Date	22-Dec-17
GIO #	GIO-104
GIO Title	Ongoing Work on Bruce B Heat Transport Vibration Project
CARD(s) Associated with this GIO	CA-0379
Gap(s) Associated with this CARD	SF02-SUP-050-16
Additional Information	In order to effectively evaluate the benefits of implementing design modifications to the PHT System, the following assessments are currently underway 1. Assessment of 7-Vane Impellor (VI) a. Flowserve (FS) design and CFD modelling of 5VI/7VI b. Acoustic/vibration modelling c. Out-reactor tests of acoustic impact of fuel Currently in Phase 1 of 7VI preliminary engineering. 2. Assessment of RISP a. Production of 6 RISPs - nearing completion of the initial 2 RISPs for out-reactor testing

	b. RISP support - preparation of I&C specifications for I&C installation in a suitable upcoming Bruce B outage in 2017 or 2018 c. Out-reactor RISP tests 3. Proposed Supporting Activities a. Submission of two bounding safety assessments of the impact of end-plate cracking (EPC) for the most limiting outer zone channel. (complete) b. Ongoing fuel inspections c. Continued re-assessment of extent/degree of PHT vibration Issues d. Periodic update of the EPC safety assessment, as required e. Application of RIDM process to Determine path forward. Note this is a IIP milestone date for the development and submission of a plan and schedule for the remaining PHT Vibration Project scope. Upon completion of milestone, the TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	NK21-CORR-00531-13357/NK29-CORR-00531-13907

CARD #	CA-0294
CARD Title	Licence Concessions Database
CARD Description	Develop and implement a controlled, centralized and accessible company database that allows for the tracking of concessions granted to Bruce Power by the CNSC.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTRA
Functional Area	DPTRA
Value Tree Tier 3 Objective	2.3.1 - Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.49160
Time Attribute Score	5
Time Attribute Rationale	Developing and implementing a controlled, centralized and accessible company database will have an immediate impact on CNSC concession management and other regulatory interfaces.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 4b- Relates to improvement of the managed system and support processes, Column 1- Developing and implementing a controlled, centralized and accessible company database will be a new practice in managing CNSC concessions and interface.
Time-Impact Utility Score	0.46118
Final Score	0.22672
CARD Priority	3
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-088
GIO Title	Improve Licencing Processes
CARD(s) Associated with this GIO	CA-0294
Gap(s) Associated with this CARD	SF01_ANSI/NIRMA CM 1.0-2007_3.2_16
Additional Information	
References	

CARD #	CA-0078
CARD Title	Improvement of unavailability targets for some safety related systems
CARD Description	Improve the probability of failure on demand from all causes for safety systems to be consistently lower than 1E-3. Standby Class III Power System predicted unavailability targets exceeded in 2012 and 2013 due to an inconsistency between the modelling and plant operation. Take action to remove the inconsistency to reduce the unavailability.
Applicable Units	Bruce A
Alert Group	DPTRSS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Resolution of the issue(s) through completion of this CA will take up to 3 years to have its effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix- Row 2, Column 2 Improving probability of failure on demand augments operational and safety performance.
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-034
GIO Title	Safety System Reliability
CARD(s) Associated with this GIO	CA-0078
Gap(s) Associated with this CARD	SF01_CSA N290.0-11_4.5-4.8_15, SF01_CSA N290.1_4.2.1.1_15, SF08_SF8 RT 2015_5.8_15
Additional Information	
References	

CARD #	CA-0168
CARD Title	Air pressure measurements in support of emission estimates
CARD Description	In order to align with CSA N288.3.4, update the documentation and procedures on systems that remove radioactive particulate matter and iodine species from airborne effluent streams to include- Air pressure measurements in support of emission estimates
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMSE
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Alignment of Performance testing of nuclear air-cleaning systems with CSA N288.3.4 will have an immediate effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved operational safety performance. Column 2-Augments/ recovers the current Performance testing of nuclear air-cleaning systems
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-082
GIO Title	Performance testing of nuclear air-cleaning systems
CARD(s) Associated with this GIO	CA-0168, CA-0169, CA-0170, CA-0171, CA-0172, CA-0173
Gap(s) Associated with this CARD	SF14_CSA N288.3.4-13_8.5_15
Additional Information	The actions for this AR will be implemented under one action plan for CA-0168 to CA-0173.
References	

CARD #	CA-0169
CARD Title	QA/QC guidance for performance testing of nuclear air-cleaning systems
CARD Description	In order to align with CSA N288.3.4, update the documentation and procedures on systems that remove radioactive particulate matter and iodine species from airborne effluent streams to include-QA/QC guidance for performance testing of nuclear air-cleaning systems
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMSE
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Alignment of Performance testing of nuclear air-cleaning systems with CSA N288.3.4 will have an immediate effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved operational safety performance. Column 2-Augments/ recovers the current Performance testing of nuclear air-cleaning systems.
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-082
GIO Title	Performance testing of nuclear air-cleaning systems
CARD(s) Associated with this GIO	CA-0168, CA-0169, CA-0170, CA-0171, CA-0172, CA-0173
Gap(s) Associated with this CARD	SF14_CSA N288.3.4-13_10_15
Additional Information	The actions for this AR will be implemented under one action plan for CA-0168 to CA-0173.
References	

CARD #	CA-0170
CARD Title	Effectiveness reviews of the air-cleaning system performance testing program
CARD Description	In order to align with CSA N288.3.4, update the documentation and procedures on systems that remove radioactive particulate matter and iodine species from airborne effluent streams to include-Effectiveness reviews of the air-cleaning system performance testing program. In addition, conduct self-assessments or independent audits of the air-cleaning system performance testing program.
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMSE
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Alignment of Performance testing of nuclear air-cleaning systems with CSA N288.3.4 will have an immediate effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved operational safety performance. Column 2-Augments/ recovers the current Performance testing of nuclear air-cleaning systems.
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-082
GIO Title	Performance testing of nuclear air-cleaning systems
CARD(s) Associated with this GIO	CA-0168, CA-0169, CA-0170, CA-0171, CA-0172, CA-0173
Gap(s) Associated with this CARD	SF14_CSA N288.3.4-13_11_15
Additional Information	The actions for this AR will be implemented under one action plan for CA-0168 to CA-0173.
References	

CARD #	CA-0171
CARD Title	Requirements for the qualifications of personnel who conduct air filter performance testing
CARD Description	In order to align with CSA N288.3.4, update the documentation and procedures on systems that remove radioactive particulate matter and iodine species from airborne effluent streams to include- Requirements for the qualifications of personnel who conduct air filter performance testing.
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMSE
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Alignment of Performance testing of nuclear air-cleaning systems with CSA N288.3.4 will have an immediate effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved operational safety performance. Column 2-Augments/ recovers the current Performance testing of nuclear air-cleaning systems.
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-082
GIO Title	Performance testing of nuclear air-cleaning systems
CARD(s) Associated with this GIO	CA-0168, CA-0169, CA-0170, CA-0171, CA-0172, CA-0173
Gap(s) Associated with this CARD	SF14_CSA N288.3.4-13_12_15
Additional Information	The actions for this AR will be implemented under one action plan for CA-0168 to CA-0173.
References	

CARD #	CA-0172
CARD Title	Performance testing of nuclear air-cleaning systems- Program documentation
CARD Description	Conduct a gap assessment and develop documentation and procedures to meet the requirements of Clause 13 of N288.3.4 for systems that remove radioactive particulate matter and iodine species from airborne effluent streams.
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMSE
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Alignment of Performance testing of nuclear air-cleaning systems with CSA N288.3.4 will have an immediate effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved operational safety performance. Column 2-Augments/ recovers the current Performance testing of nuclear air-cleaning systems.
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-082
GIO Title	Performance testing of nuclear air-cleaning systems
CARD(s) Associated with this GIO	CA-0168, CA-0169, CA-0170, CA-0171, CA-0172, CA-0173
Gap(s) Associated with this CARD	SF14_CSA N288.3.4-13_13_15
Additional Information	The actions for this AR will be implemented under one action plan for CA-0168 to CA-0173.
References	

CARD #	CA-0173
CARD Title	Pre-service and in-service testing of adsorbent media (activated carbon)
CARD Description	In order to align with CSA N288.3.4, update the documentation and procedures on systems that remove radioactive particulate matter and iodine species from airborne effluent streams to include-Pre-service and in-service testing of adsorbent media (activated carbon).
Applicable Units	Bruce A & Bruce B
Alert Group	DIVDMSE
Functional Area	DPTERI
Value Tree Tier 3 Objective	2.2.1 - Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life
Tier 3 Weight	0.32159
Time Attribute Score	5
Time Attribute Rationale	Alignment of Performance testing of nuclear air-cleaning systems with CSA N288.3.4 will have an immediate effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improved operational safety performance. Column 2-Augments/ recovers the current Performance testing of nuclear air-cleaning systems.
Time-Impact Utility Score	0.70374
Final Score	0.22632
CARD Priority	4
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-082
GIO Title	Performance testing of nuclear air-cleaning systems
CARD(s) Associated with this GIO	CA-0168, CA-0169, CA-0170, CA-0171, CA-0172, CA-0173
Gap(s) Associated with this CARD	SF14_CSA N288.3.4-13_8.9_15
Additional Information	The actions for this AR will be implemented under one action plan for CA-0168 to CA-0173.
References	

CARD #	CA-0061
CARD Title	SIP-16: BA U1/U2 Post RTS - Seismic Margin Upgrade (IIP-6)
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12647 - ACTION ITEM 1407-4602: BRUCE A SEISMIC MARGIN ASSESSMENT - SEMI-ANNUAL UPDATE #5
Applicable Units	Unit 1 & 2
Alert Group	DPTPROJA
Functional Area	DIVDMES
Value Tree Tier 3 Objective	3.1.2 - Enhanced confidence in the equipment qualification requirements resulting from hazards analysis of internal and external events
Tier 3 Weight	0.14751
Time Attribute Score	5
Time Attribute Rationale	Installation of B1&2 Seismic Margin Upgrades will have an immediate impact on improving design basis for equipment qualification
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features against seismic events, Column 1- Introduces new design features in support of seismic qualification
Time-Impact Utility Score	1.00000
Final Score	0.14750
CARD Priority	5
CARD Status	Underway
Target Completion Date	30-Mar-18
GIO #	GIO-019
GIO Title	Assess and improve seismic qualification
CARD(s) Associated with this GIO	CA-0061
Gap(s) Associated with this CARD	SF01-IIP-004-14
Additional Information	SIP-16 and AI 1407-4602 are complete. As stated in NK21-CORR-00531-13426, closure of the action item has been requested.
References	NK21-CORR-00531-13426 NK21-CORR-00531-12257 NK21-CORR-00531-12647 NK21-CORR-00531-11170 AI 1407-4602

CARD #	CA-0376
CARD Title	Development and Implementation of Whole-Site Probabilistic Risk Assessment
CARD Description	This initiative is implemented under projects 39020 & 39236 - Development of a Probabilistic Safety Program & Implementation of a Whole Site Probabilistic Safety Assessment.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTNSAS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	4.1.1 - Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record
Tier 3 Weight	0.14010
Time Attribute Score	5
Time Attribute Rationale	Development and Implementation of Whole-Site Probabilistic Risk Assessment will have immediate impact on improving the understandin of risk associated with the whole-site.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reducing uncertainty in the design basis and engineered safety features by Development and Implementation of Whole-Site Probabilistic Risk Assessment, Column 1- This is a new Probabilistic Risk Assessment and also augments the unit specific assessments.
Time-Impact Utility Score	0.70374
Final Score	0.09859
CARD Priority	6
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-089
GIO Title	Whole-Site Probabilistic Risk Assessment
CARD(s) Associated with this GIO	CA-0376
Gap(s) Associated with this CARD	SF06-SUP-046-16, SF06-SUP-047-16
Additional Information	
References	NK21-CORR-00531-11715/NK29 -CORR-00531-12105 NK21- CORR-00531-12837/NK29- CORR-00531-13287 NK21- CORR-00531-12973/NK29- CORR-00531-13444 NK21- CORR-00531-13030/NK29- CORR-00531-13499

CARD #	CA-0334
CARD Title	Control Distribution Frame (CDF) Terminal Replacement - Unit 3
CARD Description	The work covers replacement of Control Distribution Frame (CDF) Terminal blocks in a unit and common equipment rooms. The components affected are the terminal blocks, complete with endplates, clamping screws, labels and associated accessories on the CDF only.
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.2 - Enhanced confidence in maintaining SIR in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.07023
Time Attribute Score	5
Time Attribute Rationale	Unit 3- Control Distribution Frame (CDF) Terminal Replacement will have an immediate impact on maintaining the design basis CDF.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of CDF. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.07020
CARD Priority	7
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-064
GIO Title	Control Distribution Frame (CDF) Terminal Replacement
CARD(s) Associated with this GIO	CA-0334, CA-0335
Gap(s) Associated with this CARD	SF02-MCR-0130-16
Additional Information	AMOT-0181
References	

CARD #	CA-0335
CARD Title	Control Distribution Frame (CDF) Terminal Replacement - Unit 4
CARD Description	The work covers replacement of Control Distribution Frame (CDF) Terminal blocks in a unit and common equipment rooms. The components affected are the terminal blocks, complete with endplates, clamping screws, labels and associated accessories on the CDF only.
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DPTPRJSUP
Value Tree Tier 3 Objective	2.3.2 - Enhanced confidence in maintaining SIR in a state that achieves reliable operation and safety performance and extended plant life
Tier 3 Weight	0.07023
Time Attribute Score	5
Time Attribute Rationale	Unit 4- Control Distribution Frame (CDF) Terminal Replacement will have an immediate impact on maintaining the design basis CDF.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to maintaining the design basis of CDF. Column 2-Augments/ recovers the current margins in the design.
Time-Impact Utility Score	1.00000
Final Score	0.07020
CARD Priority	7
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-064
GIO Title	Control Distribution Frame (CDF) Terminal Replacement
CARD(s) Associated with this GIO	CA-0334, CA-0335
Gap(s) Associated with this CARD	SF02-MCR-0131-16
Additional Information	AMOT-0181
References	

CARD #	CA-0043
CARD Title	SIP-3:REGDOC-2.4.1 Implementation
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12534/NK29-CORR-00531-12961 - ACTION ITEM 090739: SAFETY REPORT IMPROVEMENT PROJECT - REGULATORY COMMUNICATION PLAN
Applicable Units	Bruce A & Bruce B
Alert Group	DPTNSAS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	4.1.1 - Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record
Tier 3 Weight	0.14010
Time Attribute Score	5
Time Attribute Rationale	Upgrade of Safety Report and associated Safety Analysis in compliance with RD-310 (now REGDOC-2.4.1) will have an immediate impact on enhanced confidence in safety analysis.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reduction of uncertainty in engineered safety features through improved safety analysis, Column 2- Assures compliance with modern codes and standards and augments the current analysis practices.
Time-Impact Utility Score	0.46118
Final Score	0.06461
CARD Priority	8
CARD Status	Underway
Target Completion Date	22-Dec-17
GIO #	GIO-009
GIO Title	Update safety analysis to align with REGDOC-2.4.1
CARD(s) Associated with this GIO	CA-0043, CA-0174
Gap(s) Associated with this CARD	SF05-IIP-006-14
Additional Information	Bruce Power submitted a an implementation strategy for REGDOC-2.4.1 together with a regulatory communication plan under AI 090739 in Reference NK21-CORR-00531-12334 / NK29-CORR-00531-12767, which provides the schedule for planned submissions and meetings in support of the 2017 Safety Report submission.
References	NK21-CORR-00531-12334 / NK29-CORR-00531-12767. NK21-CORR-00531-10774 NK29-CORR-00531-11155 NK21-CORR-00531-11214 NK29-CORR-00531-11621 AI 090739

CARD #	CA-0174
CARD Title	Safety Report & Probabilistic Safety Assessment
CARD Description	When updating hazard analysis in the SR, develop an integrating section to confirm completeness, to ensure the hazard assessments remain current as knowledge is improved and modifications are made to the SSCs, and the integration and overlap with Deterministic Safety Analysis and Probabilistic Safety Assessments are well known.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTNSAS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	4.1.1 - Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record
Tier 3 Weight	0.14010
Time Attribute Score	5
Time Attribute Rationale	Inclusion of an integrating section to confirm completeness of hazard analysis will have an immediate impact on enhanced confidence in safety analysis
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reduction of uncertainty in engineered safety features through improved safety analysis, Column 2- Augments current understanding by improving current practices in documenting hazard analysis
Time-Impact Utility Score	0.46118
Final Score	0.06461
CARD Priority	8
CARD Status	Not Started
Target Completion Date	23-Dec-22
GIO #	GIO-009
GIO Title	Update safety analysis to align with REGDOC-2.4.1
CARD(s) Associated with this GIO	CA-0043, CA-0174
Gap(s) Associated with this CARD	SF08_SF8 RT_5.7_16
Additional Information	
References	

CARD #	CA-0377
CARD Title	Implementation of Asset Management Activities for Safety Significant Assets
CARD Description	Provide progress reports on the Asset Management Program on a three year frequency to the CNSC. Reporting Schedule: 1. Asset Management Activities Baseline Report; TCD = 31Jan2018 2. Asset Management Activities Progress Report #1; TCD = 28Jun2019 3. Asset Management Activities Progress Report #2; TCD = 30Jun2022 4. Asset Management Activities Progress Report #3; TCD = 30Jun2025 5. Asset Management Activities Progress Report #4; TCD = 30Jun2028 Content: Summary/Status of Asset Management Activities and Related Programs for safety significant Assets • Development of Baseline for Asset Management Activities • Current status of relevant Asset Management documents (e.g., TBAs, LCMPs, and associated TLAAs) • Risks and Opportunities to Implementation Effectiveness of Asset Management Implementation (only for reports 2-4) • Completed Activities • Advanced Activities • Delayed Activities • Attachment • Schedule/Plan for Next Reporting Period This CARD will also address comments from CNSC eDoc 5229600 (NK21-CORR-00531-13581) (Table C-1 #19, #20 and Table C-2 #24, #25 and #26), regarding status of Time Limited Aging Assessments (TLAAs), as well as revisions to the suite of Technical Basis Assessments (TBAs) in support of Life Cycle Management Plans (LCMPs). The governance for TBAs and LCMPs is being revised to provide improved clarity on the criteria for determining which assets require this formal documentation of aging related strategies, and to better describe the relationship between basis documentation and the approved plans to manage the life cycle of safety related assets. These governance updates will also capture the ongoing activities both within Bruce Power and in collaboration with the COG Asset Management Peer Group, to complete the documentation of the strategies to address TLAAs applicable to Bruce Power structures and components.

	Progress on the associated deliverables for TLAAs and revised TBAs and LCMPs as part of the Bruce Power Plan-Do-Check-Act process of continuous improvement, will be included as part of the periodic progress reports.
Applicable Units	Bruce A & B
Alert Group	DPTERI
Functional Area	DIVDMSE
Value Tree Tier 3 Objective	2.1.1 - Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) for an extended plant life
Tier 3 Weight	0.05695
Time Attribute Score	5
Time Attribute Rationale	Improved Asset Management Program Progress Reporting will have an immediate impact on the robustness of Asset Management Program.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to improved understanding and documentation of the condition of SSCs through the Asset Management Program and reporting of the same to the CNSC. Column 2-Augments/ recovers the current processes, programs and reporting in place.
Time-Impact Utility Score	1.00000
Final Score	0.05700
CARD Priority	9
CARD Status	Not Started
Target Completion Date	31-Jan-18
GIO #	GIO-103
GIO Title	Implementation of Asset Management Activities
CARD(s) Associated with this GIO	CA-0377
Gap(s) Associated with this CARD	SF02-SUP-048-16
Additional Information	Note that the TCD is a milestone date. Upon completion of the milestone, the TCD will be updated with the next reporting cycle date. The IIP item will remain open until the last report has been issued.
References	

CARD #	CA-0314
CARD Title	BB Maintenance Cooling Interspace Protection
CARD Description	This initiative is implemented under Project # 36109 - Bruce B Maintenance Cooling Interspace Protection.
Applicable Units	Bruce B
Alert Group	DPTMCD
Functional Area	DPTERI
Value Tree Tier 3 Objective	1.1.1 - Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation
Tier 3 Weight	0.03498
Time Attribute Score	5
Time Attribute Rationale	Installing correctly sized maintenance cooling relief valves will have an immediate impact on preventing chattering of the existing oversized relief valves in the system. Chattering could cause a significant challenge to the D2O collection system.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.03500
CARD Priority	10
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-099
GIO Title	Install Correctly Sized Maintenance Cooling Relief Valves
CARD(s) Associated with this GIO	CA-0314
Gap(s) Associated with this CARD	SF01-SUP-017-16
Additional Information	Bruce B currently has a regulatory commitment to bring all pressure retaining systems in to compliance with licence conditions by December 31, 2017. During the assessment of the Bruce B MCS for the legacy registration project, issues were identified with the current configuration. As a result, physical changes to the system are required to support the registration of the MCS. The preferred alternative cannot be completed during 2017 to meet the regulatory commitment for the Bruce B legacy registration project. Bruce Power is therefore providing notification that the legacy registration of the MCS (34720B) will be completed as part of the MCS relief valve project that will be a commitment under the IIP, and not under the regulatory commitment for legacy registration. The proposed MCS relief valve project will include the system's legacy

	registration update, in addition to registering the MCS vent line modifications installed during recent Unit 5, Unit 7 and Unit 8 outages in accordance with variances accepted by CNSC staff in the following letters: K. Lafrenière to F. Saunders "Bruce B Unit 5 and Unit 7: Variance for Maintenance Cooling System Vent Line", August 12, 2016, NK29-CORR-00531 -13493. K. Lafrenière to F. Saunders "Bruce B Unit 8: Variance for Maintenance Cooling System Vent Line", March 10, 2016, NK29-CORR-00531-13145. Note: The TCD is a milestone date; upon completion of milestone, TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	NK29-CORR-00531-14091 NK29-CORR-00531-13950

CARD #	CA-0089
CARD Title	Validation of human actions credited under accident conditions in the safety report
CARD Description	Establish if human actions identified and credited in the Bruce A and Bruce B Safety Report were validated. Validate human actions identified and credited in the Bruce B Safety Report as required. Update relevant training and safety analysis procedures for input from training exercises, particularly those modeling accident conditions, to safety analyses to validate assumptions.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTOCP
Functional Area	DIVDMES
Value Tree Tier 3 Objective	5.2.3 - Enhanced confidence in the appropriateness of plant control interfaces (human factors)
Tier 3 Weight	0.04572
Time Attribute Score	5
Time Attribute Rationale	Resolution of the issue(s) through completion of this CA will take up to 3 years to have its effect on the objective.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix- Row 2, Column 2 Validation of human actions identified and credited in the Safety Report augments operational and safety performance.
Time-Impact Utility Score	0.70374
Final Score	0.03216
CARD Priority	11
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-043
GIO Title	Validation of Human Credited Actions
CARD(s) Associated with this GIO	CA-0089, CA-0177
Gap(s) Associated with this CARD	SF12_SF12 RT_5.4_16, SF12_SF12 RT_5.4_16, SF12_SF12 RT_5.4_15
Additional Information	
References	

CARD #	CA-0318
CARD Title	RP Instrumentation maintenance
CARD Description	There are gaps in the effective implementation of the RP instrumentation program in order to maintain the fixed RP instrumentation (specifically FAGMs and whole-body contamination monitors) in good working condition.
Applicable Units	Bruce A & B
Alert Group	DPTRPR
Functional Area	DPTERI
Value Tree Tier 3 Objective	5.3.1 - Enhanced confidence in radiation protection
Tier 3 Weight	0.04349
Time Attribute Score	5
Time Attribute Rationale	Improvement in RP Instrumentation maintenance will have an immediate impact on enhanced RP.
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Relates to improving RP Instrumentation maintenance. Column 2-Augments/recovers the current barriers in RP Instrumentation maintenance.
Time-Impact Utility Score	0.70374
Final Score	0.03061
CARD Priority	12
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-093
GIO Title	RP equipment and instrumentation maintenance and life cycle management
CARD(s) Associated with this GIO	CA-0317, CA-0318, CA-0320
Gap(s) Associated with this CARD	SF15_SF15 RT_5.2.2_15
Additional Information	
References	

CARD #	CA-0317
CARD Title	RP Instrumentation life cycle management
CARD Description	There is no documented lifecycle management process for the FAGMs system.
Applicable Units	Bruce A & B
Alert Group	DIVDMSE (DPTPEBA & DPTPEBB)
Functional Area	DPTERI
Value Tree Tier 3 Objective	5.3.1 - Enhanced confidence in radiation protection
Tier 3 Weight	0.04349
Time Attribute Score	5
Time Attribute Rationale	Improvement in RP lifecycle management process for the FAGMs system will have an immediate impact on enhanced RP.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to improving LCM for FAGMs. Column 2-Augments/recovers the current barriers in LCM for FAGMs.
Time-Impact Utility Score	0.46118
Final Score	0.02006
CARD Priority	13
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-093
GIO Title	RP equipment and instrumentation maintenance and life cycle management
CARD(s) Associated with this GIO	CA-0317, CA-0318, CA-0320
Gap(s) Associated with this CARD	SF15_SF15 RT_5.2.2_15
Additional Information	
References	

CARD #	CA-0320
CARD Title	Technical Basis for RP instrumentation setpoints, locations and function checks
CARD Description	Technical Basis for whole-body monitor alarm test frequency is not available in formal documentation
Applicable Units	Bruce A & B
Alert Group	DPTRPR
Functional Area	DPTERI
Value Tree Tier 3 Objective	5.3.1 - Enhanced confidence in radiation protection
Tier 3 Weight	0.04349
Time Attribute Score	5
Time Attribute Rationale	Improvement in technical Basis for RP instrumentation setpoints, locations and function checks will have an immediate impact on enhanced RP.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to improving technical basis for RP test procedures. Column 2-Augments/recovers the current barriers in RP test procedures.
Time-Impact Utility Score	0.46118
Final Score	0.02006
CARD Priority	13
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-093
GIO Title	RP equipment and instrumentation maintenance and life cycle management
CARD(s) Associated with this GIO	CA-0317, CA-0318, CA-0320
Gap(s) Associated with this CARD	SF15_WANO GL 2004-01-R1_VI.C2._15, SF15_WANO GL 2004-01-R1_VI.C2._15, SF15_WANO GL 2004-01-R1_VI.C2._15, SF15_WANO GL 2004-01-R1_VI.C2._15, SF15_WANO GL 2004-01-R1_VI.C2._15, SF15_WANO GL 2004-01-R1_VI.C2._15
Additional Information	
References	

CARD #	CA-0319
CARD Title	Improve effective use of the action tracking system in Radiation Protection
CARD Description	Develop process for the effective and timely dispositioning of DCRs and communicate process to RP Document Authors.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTRPR
Functional Area	DPTRPR
Value Tree Tier 3 Objective	5.3.1 - Enhanced confidence in radiation protection
Tier 3 Weight	0.04349
Time Attribute Score	5
Time Attribute Rationale	Improvement in effective use of the action tracking system will have an immediate impact on enhanced RP.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 4a- Relates to improving RP procedures. Column 2-Augments/recovers the current barriers in RP procedure updates.
Time-Impact Utility Score	0.46118
Final Score	0.02006
CARD Priority	13
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-094
GIO Title	Effective use of the action tracking system in Radiation Protection
CARD(s) Associated with this GIO	CA-0319
Gap(s) Associated with this CARD	SF15_SF15 RT_5.4.1_15
Additional Information	
References	

CARD #	CA-0201
CARD Title	Address the following issues identified as part of the OSART review with respect to ERP
CARD Description	Address the following issues identified as part of the OSART review with respect to ERP: - confirming worker safety should parts of the plant become uninhabitable following a four unit severe accident. - the number of electronic personal dosimeters dedicated to emergency response personnel. - potential delays in obtaining personal protective equipment from stores if access is impeded. - lack of severe accident dispersion calculations; and potential errors from the use of manual accounting method for centre of site staff during emergencies or site evacuation. -radiation protection for EMC staff may not be sufficient. There is lack of a filtered ventilation system, and although the EMC can be relocated, this may delay the emergency response. -there is lack of specific public address system announcements for multi-unit severe accidents. - procedural guidance for shift managers to prioritize emergency classification could potentially lead to a delay in classifying an emergency and off-site notification.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTETC
Functional Area	DPTPRG
Value Tree Tier 3 Objective	5.4.3 - Enhanced confidence in performance monitoring and corrective action
Tier 3 Weight	0.01822
Time Attribute Score	5
Time Attribute Rationale	Addressing OSART issues will have an immediate impact on ERP effectiveness.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1, Column 2 Improvement initiatives contain a number infrastructure improvements that augment the current barriers and practices.
Time-Impact Utility Score	1.00000
Final Score	0.01820
CARD Priority	14
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-044
GIO Title	Emergency preparedness
CARD(s) Associated with this GIO	CA-0090, CA-0199, CA-0200, CA-0201

Gap(s) Associated with this CARD	SF13_SF13 RT 2016_5.3.1_16, SF13_SF13 RT 2016_5.3.1_16, SF13_SF13 RT 2016_5.3.1_16, SF13_SF13 RT 2016_5.3.1_16, SF13_SF13 RT 2016_5.3.2_16, SF13_SF13 RT 2016_5.3.3_16, SF13_SF13 RT 2016_5.3.3_16
Additional Information	
References	

CARD #	CA-0006
CARD Title	SIP-13B: BB Legacy Registration
CARD Description	Per CNSC Correspondence: NK29-CORR-00531-13701 – Action Item 091413: Bruce B Legacy Registration Project Update
Applicable Units	Bruce B
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.1.1 - Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation
Tier 3 Weight	0.03498
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to improvement of the design documentation and its configuration management, Column 2- Completion of legacy registration augments effectiveness of the current practices.
Time-Impact Utility Score	0.46118
Final Score	0.01614
CARD Priority	15
CARD Status	Underway
Target Completion Date	31-Dec-17
GIO #	GIO-001
GIO Title	Improve documented design basis
CARD(s) Associated with this GIO	CA-0006, CA-0191
Gap(s) Associated with this CARD	SF01-IIP-007-14
Additional Information	Action Item 091413 is in progress. Bruce Power is focusing on the design specifications for the nuclear systems and associated work impacting the specs. Six of the 58 design specs have been completed while 11 are in progress. Progress is also being made on the overpressure protection reports and respective design reports for nuclear systems. The containment boundary assessments and updates to SCLs have been completed. Furthermore, legacy registration issues continue to be addressed as part of engineering change control. Bruce Power confirmed it's on track to meet the December 2017 due date.
References	NK29-CORR-00531-13701 NK29-CORR-00531-12884 NK29-CORR-00531-11687 AI 091413

CARD #	CA-0191
CARD Title	Update governing procedures and implementing documents on seismic qualification
CARD Description	Update DPT-PDE-00017 and its implementing documents NK29-DG-03650-002 to reflect the latest requirements of CSA N289.1 (i.e., the 10-4 requirement for the definition of the DBE), including the 2014 update and the results of the Probabilistic Seismic Hazard Assessment done in 2011. Specifically take the following actions: -Perform a gap analysis of DPT-PDE-00017 versus N289.1; revise the procedure as applicable. - Include DPT-PDE-00017 in the governance project and track procedure update
Applicable Units	Bruce A & Bruce B
Alert Group	DPTEP
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.1.1 - Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation
Tier 3 Weight	0.03498
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to improvement of the design documentation and its configuration management, Column 2- Update of governing procedures and implementing documents on seismic qualification augments effectiveness of the current practices.
Time-Impact Utility Score	0.46118
Final Score	0.01614
CARD Priority	15
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-001
GIO Title	Improve documented design basis
CARD(s) Associated with this GIO	CA-0006, CA-0191
Gap(s) Associated with this CARD	SF01_CSA N289.1_3.1_16, SF03_CSA N289.1_5_16
Additional Information	This will be implemented by PSE-OMA-81406 project.
References	

CARD #	CA-0200
CARD Title	Addressing outstanding follow-up actions from Audits on Emergency Preparedness
CARD Description	Address outstanding follow-up actions from Audits on Emergency Preparedness: -Mutual Assist Response Team (MART) response timing - Completion of outstanding corrective actions from AU-2014-00005 - Recurring problem with staff selection for the ERO organizationERO staff selection-Audit findings from AU-2014-00005 and issues identified in self-assessment SA-TRGD-2014-06. -ERO Drill participation rate
Applicable Units	Bruce A & Bruce B
Alert Group	DPTETC
Functional Area	DPTPRG
Value Tree Tier 3 Objective	5.4.3 - Enhanced confidence in performance monitoring and corrective action
Tier 3 Weight	0.01822
Time Attribute Score	5
Time Attribute Rationale	Addressing outstanding follow-up actions from Audits on Emergency Preparedness will have an immediate impact on ERP effectiveness.
Impact Attribute Score	2
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2, Column 2- Address outstanding follow-up actions from Audits will improve response capability to emergencies and augment the current barriers and practices.
Time-Impact Utility Score	0.26259
Final Score	0.00478
CARD Priority	16
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-044
GIO Title	Emergency preparedness
CARD(s) Associated with this GIO	CA-0090, CA-0199, CA-0200, CA-0201
Gap(s) Associated with this CARD	SF13_SF13 RT 2015_7.4_15, SF13_SF13 RT 2016_5.1_16, SF13_SF13 RT 2016_7.1_16, SF13_SF13 RT 2016_7.2.1.1_16, SF13_SF13 RT 2016_7.4_16
Additional Information	
References	

CARD #	CA-0047
CARD Title	SIP-11: Fukushima Response - Severe Accident Management Enhancements
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 - BRUCE POWER PROGRESS REPORT NO. 9 ON CNSC ACTION PLAN - FUKUSHIMA ACTION ITEMS
Applicable Units	Bruce A & Bruce B
Alert Group	DPTPROJC
Functional Area	DIVDMES
Value Tree Tier 3 Objective	6.2.2 - Enhanced confidence in the ability to mitigate releases associated with beyond design basis events
Tier 3 Weight	0.00560
Time Attribute Score	5
Time Attribute Rationale	Severe Accident Management Enhancements as a follow-up to Fukushima will have an immediate impact on enhanced confidence in the ability to mitigate releases associated with severe accidents
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reducing uncertainty in engineered safety features, Column 1- Assures SAMG addresses multi-unit and IFB events as well as assessment of multi-unit and IFB events, plant habitability and instrument/equipment survivability which was not covered previously
Time-Impact Utility Score	0.70374
Final Score	0.00394
CARD Priority	17
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-011
GIO Title	Implement enhancements to SAMG
CARD(s) Associated with this GIO	CA-0047
Gap(s) Associated with this CARD	SF05-IIP-008-14
Additional Information	SIP-11 action is in progress. The SAMG updates to address multi-unit events and irradiated fuel bay events have been completed as described in NK21-CORR-00531-12209 / NK29-CORR-00531-12635. The station specific assessment for instrument / equipment survivability and habitability were provided in NK21-CORR-00531-11801 / NK29-CORR-00531-12195. Closure is pending the completion of the related modifications for SAMG implementation. The progress on implementing the modifications, being tracked under AI 2014-07-3688, is described in Attachment B, Sections 2.1 and 2.3 of NK21 -CORR-00531 -12554 / NK29-CORR-00531 -12979 / NK37-CORR-00531 -

	02511. Installation to be executed during unit outages starting Q1 2017. Installation to be executed during unit outages starting Q1 2017 as per NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560. Individual outages are as follows: Unit 1: Q2 2018 Unit 2: Q2 2019 Unit 3: Q4 2018 Unit 4: Q3 2018 Unit 5: Q1 2017 Unit 6: Q3 2017 Unit 7: Q2 2019 Unit 8: Q4 2018 Total of 8 SAMG kits (TMOD) for Secondary flow path (1 for each unit) Bruce A available by Q3 2016 Bruce B available by Q4 2016
References	NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437. NK21-CORR-00531-11801 / NK29-CORR-00531-12195 / NK37-CORR-00531-02338. NK21 -CORR-00531 -12554 / NK29-CORR-00531 -12979 / NK37-CORR-00531 -02511 2014-07-3688 NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254

CARD #	CA-0177
CARD Title	Definition of staff availability requirements for supporting heat sink availability
CARD Description	Update operating documentation to explicitly require the staff credited with performing contingency activities to support the heat sink not to be credited with availability for other activities.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTOCP
Functional Area	DPTOCP
Value Tree Tier 3 Objective	5.2.1 - Enhanced confidence in the comprehensiveness and effectiveness of procedures
Tier 3 Weight	0.00524
Time Attribute Score	5
Time Attribute Rationale	Definition of staff requirements for supporting heat sinks will have an immediate impact in heat sink availability
Impact Attribute Score	4
Impact Attribute Rationale	Impact Evaluation Matrix: Row 2- Definition of staff availability requirements for supporting heat sink availability improves operational safety performance, Column 2- Definition of staff availability requirements for supporting heat sink availability will improve operational and safety performance and augment the current barriers and practices.
Time-Impact Utility Score	0.70374
Final Score	0.00366
CARD Priority	18
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-043
GIO Title	Validation of Human Credited Actions
CARD(s) Associated with this GIO	CA-0089, CA-0177
Gap(s) Associated with this CARD	SF01_CSA N290.11-13_5.2.2.4_15, SF01_CSA N290.11-13_5.2.2.4_16
Additional Information	
References	

CARD #	CA-0192
CARD Title	SF1-3: Perform an assessment of pipe whip and jet impingement
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12191 - CLOSED ACTION ITEM 090732: ASSESSMENT OF UNITS 1 AND 2 PIPE WHIP AND JET IMPINGEMENT Tracked under Project Eng 39323 - Bruce B Assessment of Pipe Whip/Jet Impingement.
Applicable Units	Bruce B
Alert Group	DPTMCD (SECRD)
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reduction in uncertainty in engineered safety features , Column 2- Assessment will confirm compliance with complementary requirements of modern codes and standards as applied to pipe whip and jet impingement.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-003
GIO Title	Assess pipe whip and jet impingement
CARD(s) Associated with this GIO	CA-0192
Gap(s) Associated with this CARD	SF01-IIP-009-14
Additional Information	Resources have been assigned to a Bruce B project, which is expected to be kicked-off in Q1 of 2016. This will be tracked under AR 28465760.
References	NK21-CORR-00531-12191 NK21-CORR-00531-11567 / NK29-CORR-00531-11950/ NK37-CORR-00531-02288. NK21-CORR-00531-08706

CARD #	CA-0028
CARD Title	SF1-8: Evaluate impact of fatigue due to cyclic operation transient loads on Class 4 Containment Penetrations
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457 - INTEGRATED IMPLEMENTATION PLAN FOR BRUCE A, BRUCE B AND CENTRE OF SITE. Tracked under Project Eng-39324 - Bruce B Assessment of Class 2, 3,4 & 6 Systems.
Applicable Units	Bruce B
Alert Group	DPTMCD (SECRD)
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reduction in uncertainty in engineered safety features, Column 2- Assessment will confirm compliance with complementary requirements of modern codes and standards as applied to addressing impact of fatigue due cyclic operation transient loads.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-005
GIO Title	Assess cyclic loads of pressure retaining components designed per ASME III or VIII
CARD(s) Associated with this GIO	CA-0028, CA-0029, CA-0030
Gap(s) Associated with this CARD	SF01-IIP-010-14
Additional Information	This SIP action is in progress. The scope of this work is being included in the project raised to assess pipe whip and jet impingement at Bruce B. This is being tracked under AR 28465772.
References	NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457

CARD #	CA-0029
CARD Title	SF1-9: Evaluate impact of fatigue for Class 2, 3 and 4 bellows expansion joints
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457 - INTEGRATED IMPLEMENTATION PLAN FOR BRUCE A, BRUCE B AND CENTRE OF SITE Tracked under Project Eng-39324 - Bruce B Assessment of Class 2, 3,4 & 6 Systems.
Applicable Units	Bruce B
Alert Group	DPTMCD (SECRD)
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reduction in uncertainty in engineered safety features, Column 2- Assessment will confirm compliance with complementary requirements of modern codes and standards as applied to addressing impact of fatigue due cyclic operation transient loads.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-005
GIO Title	Assess cyclic loads of pressure retaining components designed per ASME III or VIII
CARD(s) Associated with this GIO	CA-0028, CA-0029, CA-0030
Gap(s) Associated with this CARD	SF01-IIP-011-14
Additional Information	This SIP action is in progress. The scope of this work is being included in the project raised to assess pipe whip and jet impingement at Bruce B. This is tracked under AR 28465779.
References	NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457

CARD #	CA-0030
CARD Title	SF1-12: Evaluate Class 6 piping components for cyclic and dynamic reactions
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457 - INTEGRATED IMPLEMENTATION PLAN FOR BRUCE A, BRUCE B AND CENTRE OF SITE Tracked under to Project Eng-39324 - Bruce B Assessment of Class 2, 3,4 & 6 Systems.
Applicable Units	Bruce B
Alert Group	DPTMCD (SECRD)
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to reduction in uncertainty in engineered safety features, Column 2- Assessment will confirm compliance with complementary requirements of modern codes and standards as applied to addressing impact of fatigue due cyclic operation transient loads and dynamic reactions.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-005
GIO Title	Assess cyclic loads of pressure retaining components designed per ASME III or VIII
CARD(s) Associated with this GIO	CA-0028, CA-0029, CA-0030
Gap(s) Associated with this CARD	SF01-IIP-012-14
Additional Information	This SIP action is in progress. The scope of this work is being included in the project raised to assess pipe whip and jet impingement at Bruce B. This is being tracked under AR 28465765.
References	NK21-CORR-00531-12288 / NK29-CORR-00531-12719 / NK37-CORR-00531-02457

CARD #	CA-0080
CARD Title	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
CARD Description	Investigate and define as appropriate (a) limits specified on mechanical vibration protection qualification of SDS equipment, and (b) protection against electromagnetic noise disturbances of SDS equipment.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTEP
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Resolution of the issue(s) through completion of this CA will take up to 3 years to have its effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix- Row 3, Relates to reduction in uncertainty in engineered safety features, Column 2- Investigation of mechanical vibration protection qualification and electromagnetic noise disturbances of SDS equipment will augment the current understanding of robustness of SDS equipment.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-036
GIO Title	Electromagnetic Noise Disturbances and Mechanical Vibration Protection Qualification of SDS Equipment
CARD(s) Associated with this GIO	CA-0080
Gap(s) Associated with this CARD	SF01_CSA N290.1_4.7.2_15, SF01_CSA N290.1_4.7.2_16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note: The TCD is a milestone date; upon completion of milestone, TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0081
CARD Title	Establish technical basis for radiation zone designation
CARD Description	Document the basis for station zoning for normal operations including consideration of the predicted dose rates or anticipated airborne radionuclides in the areas. Describe the technical basis for zone boundaries in the design documentation. In Part 2 of the Safety Report Section 12.3.3 and design documentation include predicted dose rates or airborne radionuclides as well as the criteria and rationale for radiation zone designations – including zone boundaries for accident conditions.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTEP
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Resolution of the issue(s) through completion of this CA will take up to 3 years to have its effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix- Row 3, Relates to reduction in uncertainty in engineered safety features, Column 2 Establishing technical basis for radiation zone designation will augment the current understanding of plant configuration as related to radiation zones.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-037
GIO Title	Document design basis for zoning and shielding
CARD(s) Associated with this GIO	CA-0081, CA-0082
Gap(s) Associated with this CARD	SF01_CNSC REGDOC 2.5.2_8.13_15, SF01_CNSC REGDOC 2.5.2_8.13_16, SF01_CNSC REGDOC 2.5.2_8.13.1_16, SF01_CNSC REGDOC 2.5.2_8.13.1_15
Additional Information	This will be implemented by PSE-OMA-81406 project. Note: The TCD is a milestone date; upon completion of milestone, TCD will be updated with implementation date. IIP item will remain open until

	implementation has been confirmed.
References	

CARD #	CA-0082
CARD Title	Shielding design criteria and the methodology for specification of shielding parameters and material selection
CARD Description	Update design documentation for shielding to describe: 1. shielding design criteria and the methodology for shield parameters and choice of shield material, 2. Reflect the buildup of radioactive materials over the lifetime of the NPP in the shielding specifications.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTEP
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Resolution of the issue(s) through completion of this CA will take up to 3 years to have its effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix- Row 3, Relates to reduction in uncertainty in engineered safety features, Column 2 Documentation of shielding design criteria and the methodology for specification of shielding parameters and material selection will augment the current understanding of robustness of shielding installed in the plant.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-037
GIO Title	Document design basis for zoning and shielding
CARD(s) Associated with this GIO	CA-0081, CA-0082
Gap(s) Associated with this CARD	SF01_CNSC REGDOC 2.5.2_8.13.1_16, SF01_CNSC REGDOC 2.5.2_8.13.1_15
Additional Information	This will be implemented by PSE-OMA-81406 project. Note: The TCD is a milestone date; upon completion of milestone, TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0179
CARD Title	HF design review of control room, workstations, computer interfaces, alarms systems, soft control systems, communication systems and field components relevant to safety
CARD Description	Update Bruce Power procedures related to Human Factors to include: - Guidelines for CRT based displays. - Investigate if Bruce Power procedures related to Human Factors include applicable standard(s) or guideline(s) for the soft control systems. Update Bruce Power procedures related to Human Factors to include applicable standard(s) or guideline(s) for the soft control systems. - Investigate if Bruce Power procedures related to Human Factors include applicable standard(s) or guideline(s) for the existing Computer Interfaces. Update Bruce Power procedures related to Human Factors to include applicable standard(s) or guideline(s) for Computer Interfaces (e.g. NUREG-700).
Applicable Units	Bruce A & Bruce B
Alert Group	DPTICE
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.2 - Enhanced confidence that the design specification/analysis/qualification of the plant SSCs meet the enhanced or new specification/analytical/qualification requirements included in the modern codes and standards
Tier 3 Weight	0.00583
Time Attribute Score	5
Time Attribute Rationale	Alignment with CSA N290.12 by updating the design documentation and procedures on Human Factors will have an immediate effect on planning for HF in design
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3, Relates to reduction in uncertainty in engineered safety features, Column 2- Alignment with CSA N290.12 by updating the design documentation and procedures will improve HF in design activities and augment the current barriers and practices.
Time-Impact Utility Score	0.46118
Final Score	0.00267
CARD Priority	19
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-081
GIO Title	Human Factors in Design of Nuclear Power Plants
CARD(s) Associated with this GIO	CA-0179

Gap(s) Associated with this CARD	SF12_NUREG-0700_Part_I_15, SF12_NUREG-0700_Part_I_16, SF12_NUREG-0700_Part_II_7_16, SF12_NUREG-0700_Part_II_7_16, SF12_NUREG-0700_Part_II_7_15, SF12_NUREG-0700_Part_II_7_15
Additional Information	
References	

CARD #	CA-0090
CARD Title	Emergency response documentation
CARD Description	In the development and content of nuclear emergency recovery plans- • Establish a validation process for Emergency Mitigating Equipment (EME) guidance, and execution of key operator actions during emergency exercises. • Include the following requirements as related to fitness for duty: o Arrangements shall be made to ensure that emergency workers are, to the extent practicable, designated in advance and are fit for the intended duty. o These arrangements shall include health surveillance for emergency workers for the purpose of assessing their initial fitness and continuing fitness for their intended duties • Include requirements to ensure that arrangements are in place for the protection of emergency workers and protection of helpers in an emergency for the range of anticipated hazardous conditions in which they might have to perform response functions. These arrangements, as a minimum, shall include: (a) Training those emergency workers designated as such in advance; (b) Providing emergency workers not designated in advance and helpers in an emergency immediately before the conduct of their specified duties with instructions on how to perform the duties under emergency conditions ('just in time' training); (c) Managing, controlling and recording the doses received; (d) Provision of appropriate specialized protective equipment and monitoring equipment; (e) Provision of iodine thyroid blocking, as appropriate, if exposure due to radioactive iodine is possible; (f) Obtaining informed consent to perform specified duties, when appropriate; (g) Medical examination, longer term medical actions and psychological counseling, as appropriate. • Include a requirement for the operating organization and response organizations to ensure that all practicable means are used to minimize exposures of emergency workers and helpers in an emergency in the response to a nuclear or radiological emergency (see paragraph I.2 of Appendix I of IAEA GSR Part 7), and to optimize their protection. • Provide clear criteria for authorizing exceeding dose limits in the Emergency Preparedness Program and supporting documentation. • Include obtaining qualified medical advice before any further occupational exposure occurs if an emergency worker has received an effective dose exceeding 200 mSv, or at the request of the emergency worker. • Include minimum staffing requirements such that an appropriate number of suitably qualified personnel are available to manage an emergency response at all facilities if each of the facilities is under emergency conditions simultaneously (see paragraph 5.4 of IAEA GSR Part 7).
Applicable Units	Bruce A & Bruce B
Alert Group	DPTETC
Functional Area	DPTPRG
Value Tree Tier 3	5.2.1 - Enhanced confidence in the comprehensiveness and effectiveness of

Objective	procedures
Tier 3 Weight	0.00524
Time Attribute Score	5
Time Attribute Rationale	Improving nuclear emergency recovery plans will have an immediate impact on ERP effectiveness.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix- Row 2, Improves operational safety performance, Column 2 Revisions to Emergency Response documentation to address additional requirements of REGDOC-2.10.1, IAEA GSR Part 7 and CSA N1600 and audit findings will augment and improve emergency response capability.
Time-Impact Utility Score	0.46118
Final Score	0.00240
CARD Priority	20
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-044
GIO Title	Emergency preparedness
CARD(s) Associated with this GIO	CA-0090, CA-0199, CA-0200, CA-0201
Gap(s) Associated with this CARD	SF13_IAEA GSR Part 7_5.49_16, SF13_IAEA GSR Part 7_5.52_16, SF13_IAEA GSR Part 7_5.53_16, SF13_IAEA GSR Part 7_5.57_16, SF13_IAEA GSR Part 7_5.60_16, SF13_IAEA GSR Part 7_6.11_16, SF13_CNSC REGDOC 2.10.1_2.2.4_15, SF13_CNSC REGDOC 2.10.1_2.2.6_15, SF13_CNSC REGDOC 2.10.1_2.2.6_16, SF13_CNSC REGDOC 2.10.1_2.2.8_16, SF13_CSA N1600-14_4.2.3_16, SF13_CSA N1600-14_4.5.2_16, SF13_CSA N1600-14_4.5.12_16, SF13_CSA N1600-14_5.4_16, SF13_CSA N1600-14_4.2.3_15, SF13_CSA N1600-14_4.5.2_15, SF13_CSA N1600-14_4.5.12_15, SF13_CSA N1600-14_5.4_15
Additional Information	
References	

CARD #	CA-0199
CARD Title	Complete the On-Site/Off-Site Emergency Response Communications Project
CARD Description	Complete the On-Site/Off-Site Emergency Response Communications Project to ensure that two independent means of communication are available to all emergency centres. This initiative is implemented under Project # 38485 - Fukushima Response - On-site/Off-site Communications.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTETC
Functional Area	DPTPRG
Value Tree Tier 3 Objective	5.4.1 - Enhanced confidence in management system structure, and processes and supporting infrastructure
Tier 3 Weight	0.00221
Time Attribute Score	5
Time Attribute Rationale	Completion of the On-Site/Off-Site Emergency Response Communications Project will have an immediate impact on ERP effectiveness.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1, Improves the design basis of the plant as related to On-Site/Off-Site Emergency Response Communications Column 2- Completion of the Project will augment the current barriers and practices as related to On-Site/Off-Site Emergency Response Communications.
Time-Impact Utility Score	1.00000
Final Score	0.00220
CARD Priority	21
CARD Status	Not Started
Target Completion Date	20-Dec-19
GIO #	GIO-044
GIO Title	Emergency preparedness
CARD(s) Associated with this GIO	CA-0090, CA-0199, CA-0200, CA-0201
Gap(s) Associated with this CARD	SF13_SF13 RT 2016_5.1_16
Additional Information	
References	

CARD #	CA-0190
CARD Title	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
CARD Description	-Assess the need for a specialized shoreline dispersion model of Lake Huron -Document justification for the atmospheric dispersion model selected for the dose calculations in Safety Report Part 3
Applicable Units	Bruce A & Bruce B
Alert Group	DPTNSAS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	4.2.3 - Enhanced confidence in the application of modern methodologies and criteria in the conduct of safety analysis
Tier 3 Weight	0.00359
Time Attribute Score	5
Time Attribute Rationale	Improvements to shoreline and atmospheric dispersion models will have an immediate impact on enhanced confidence in safety analysis
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3 Reduces uncertainty in engineered safety features through improved engineering analysis, Column 2 Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2 will enhance safety analysis capability and augment the current practices.
Time-Impact Utility Score	0.46118
Final Score	0.00166
CARD Priority	22
CARD Status	Not Started
Target Completion Date	18-Dec-26
GIO #	GIO-083
GIO Title	Improvements to shoreline and atmospheric dispersion models to align with CSA-N288.2
CARD(s) Associated with this GIO	CA-0190
Gap(s) Associated with this CARD	SF05_CSA N288.2_6.4.1.1_16, SF05_CSA N288.2_6.5.1.1_16
Additional Information	
References	

CARD #	CA-0009
CARD Title	SIP-1A: Fukushima Response - Bruce A External Water Makeup to Heat Transport System and Moderator System
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 - BRUCE POWER PROGRESS REPORT NO. 9 ON CNSC ACTION PLAN - FUKUSHIMA ACTION ITEMS
Applicable Units	Bruce A
Alert Group	DPTPROJC
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design basis, Column 1- External Water Makeup to Heat Transport System and Moderator System is a new barrier as an additional heat sink.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-002
GIO Title	Implement design changes to improve severe accident response
CARD(s) Associated with this GIO	CA-0009, CA-0010, CA-0011, CA-0012, CA-0013
Gap(s) Associated with this CARD	SF01-IIP-014-14
Additional Information	SIP-1A and AI 2014-07-3688 is in progress. NK21-CORR-00531-12828 was the latest update on Fukushima actions. AI 2014-07-3688 was raised in NK21-CORR-00531-11298 to track the implementation of the schedule for the design and installation of External Water Makeup to Heat Transport System and Moderator System and Shield Tank Overpressure Protection (STOP). The conceptual engineering phase for both the Bruce A and the Bruce B STOP has been completed. Installation is planned during unit outages starting in Q2 2018. Detailed schedule for STOP, is provided in Attachment B, Table B1 of NK21-CORR-00531-12828 / NK29-CORR-00531-13279 /NK37-CORR-00531-02560. Targeted Outages: Unit 1: Q2 2018

	Unit 2: Q2 2019 Unit 3: Q4 2018 Unit 4: Q3 2018
References	NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-11298 / NK29-CORR-00531-11708 / NK37-CORR-00531-02229. NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 AI 2014-07-3688

CARD #	CA-0010
CARD Title	SIP-1B: Fukushima Response - Bruce B External Water Makeup to Heat Transport System and Moderator System
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 - BRUCE POWER PROGRESS REPORT NO. 9 ON CNSC ACTION PLAN - FUKUSHIMA ACTION ITEMS
Applicable Units	Bruce B
Alert Group	DPTPROJC
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design basis, Column 1- External Water Makeup to Heat Transport System and Moderator System is a new barrier as an additional heat sink.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-002
GIO Title	Implement design changes to improve severe accident response
CARD(s) Associated with this GIO	CA-0009, CA-0010, CA-0011, CA-0012, CA-0013
Gap(s) Associated with this CARD	SF01-IIP-013-14
Additional Information	SIP-1B and AI 2014-07-3688 are in progress NK29-CORR-00531-13279 was the latest update on Fukushima actions. These semi-annual updates will continue to be provided to the CNSC. AI 2014-07-3688 was raised in NK21-CORR-00531-11298 to track the implementation of the schedule for the design and installation of External Water Makeup to Heat Transport System and Moderator System and Shield Tank Overpressure Protection (STOP). The conceptual engineering phase for both the Bruce A and the Bruce B STOP has been completed. Installation is planned during unit outages starting in Q2 2019. Detailed schedule for STOP, HTS Make-up, Moderator Make-up and Shield Tank Make-up is provided in Attachment B, Table B1 of NK29-CORR-00531-13279. Targeted Outages: Unit 5: Q1 2017

	Unit 6: Q3 2017 Unit 7: Q2 2019 Unit 8: Q4 2018
References	NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-11298 / NK29-CORR-00531-11708 / NK37-CORR-00531-02229. NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 AI 2014-07-3688

CARD #	CA-0011
CARD Title	SIP-2A: Fukushima Response - Bruce A Containment Venting Connection Point and Passive CFVS Installation
CARD Description	Per Report: B-REP-34310-00002 (CONCEPTUAL ENGINEERING REPORT - PASSIVE CONTAINMENT FILTERED VENTING SYSTEM FUKUSHIMA MODIFICATION)
Applicable Units	Bruce A
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design basis, Column 1-Containment Venting Connection Point is a new barrier which facilitates installation of CFVS.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	30-Mar-18
GIO #	GIO-002
GIO Title	Implement design changes to improve severe accident response
CARD(s) Associated with this GIO	CA-0009, CA-0010, CA-0011, CA-0012, CA-0013
Gap(s) Associated with this CARD	SF01-IIP-016-14
Additional Information	SIP-2A actions are in progress. An assessment of options for ensuring containment integrity and filtered venting in the event of a multi-unit severe accident has concluded that existing design capability and emergency mitigation measures are a viable alternative to the installation of a filter vent system. However, supplementary evaluations of improvements that strengthen defence-in-depth showed that the best option to maintain containment integrity is a passive containment filtered venting system (CFVS). Bruce Power report B-REP-34310-00002 showed that the AREVA Dry Filtered Method was the preferred option for passive CFVS, retrofitted to Transfer Chamber 6. In the interim, Bruce Power has installed connection points for a possible future system at both Bruce A and Bruce B.

	Note that the TCD is a milestone date. Upon completion of the milestone, the TCD will be updated with the implementation date. The IIP item will remain open until implementation has been confirmed.
References	B-REP-34310-00002 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-12417 / NK29-CORR-00531-12829 / NK37-CORR-00531-02474. AI 2015-07-3683 NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254

CARD #	CA-0012
CARD Title	SIP-2B: Fukushima Response Bruce B - Containment Venting Connection Point and Passive CFVS Installation
CARD Description	Per Report: B-REP-34310-00002 (CONCEPTUAL ENGINEERING REPORT - PASSIVE CONTAINMENT FILTERED VENTING SYSTEM FUKUSHIMA MODIFICATION
Applicable Units	Bruce B
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design basis, Column 1-Containment Venting Connection Point is a new barrier which facilitates installation of CFVS.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	30-Mar-18
GIO #	GIO-002
GIO Title	Implement design changes to improve severe accident response
CARD(s) Associated with this GIO	CA-0009, CA-0010, CA-0011, CA-0012, CA-0013
Gap(s) Associated with this CARD	SF01-IIP-015-14
Additional Information	SIP-2B actions are in progress. An assessment of options for ensuring containment integrity and filtered venting in the event of a multi-unit severe accident has concluded that existing design capability and emergency mitigation measures are a viable alternative to the installation of a filter vent system. However, supplementary evaluations of improvements that strengthen defence-in-depth showed that the best option to maintain containment integrity is a passive containment filtered venting system (CFVS). Bruce Power report B-REP-34310-00002 showed that the AREVA Dry Filtered Method was the preferred option for passive CFVS, retrofitted to Transfer Chamber 6. In the interim, Bruce Power has installed connection points for a possible future system at both Bruce A and Bruce B.

	Note that the TCD is a milestone date. Upon completion of the milestone, the TCD will be updated with the implementation date. The IIP item will remain open until implementation has been confirmed.
References	B-REP-34310-00002 NK21-CORR-00531-12209 / NK29-CORR-00531-12635 / NK37-CORR-00531-02437 NK21-CORR-00531-12417 / NK29-CORR-00531-12829 / NK37-CORR-00531-02474NK21-CORR-00531-12828 / NK29-CORR-00531-13279 / NK37-CORR-00531-02560 AI 2015-07-3683 NK21-CORR-00531-11379 NK29-CORR-00531-11782 NK37-CORR-00531-02254 B-REP-34310-00002

CARD #	CA-0013
CARD Title	SIP-4: Fukushima Response (SAMG Improvement) - Bruce A Wide range ECI Sump Level Indication
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12123 - BRUCE A: EMERGENCY COOLANT INJECTION RECOVERY SUMPLEVEL MONITORING TUBING PENETRATIONS This initiative is implemented under Project # 37598 - BA Fukushima Response - ECI Recovery Sump Level Monitoring Loop Mods.
Applicable Units	Bruce A
Alert Group	DPTPROJC
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design basis, Column 1-Bruce A Wide range ECI Sump Level Indication is a new barrier which improves effectiveness of the mitigating actions to prevent containment breach under severe accident conditions.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	21-Dec-18
GIO #	GIO-002
GIO Title	Implement design changes to improve severe accident response
CARD(s) Associated with this GIO	CA-0009, CA-0010, CA-0011, CA-0012, CA-0013
Gap(s) Associated with this CARD	SF01-IIP-017-14
Additional Information	SIP-4 action is in progress. NK21-CORR-00531-12282 is the latest update on this modification, in which the CNSC approved the modification to Bruce A, in accordance with PROL 18.00/2020, Licence Condition 5.2 and the Licence Conditions Handbook, LCH-BNGS-R000, Section 5.2: To support the response to a Beyond Design Basis Accident resulting in a loss of heat sink, Bruce Power is increasing the water level indication range in the Fuelling Machine Duct ECI sump. This requires that the instrument reference lines be rerouted through Airlock OA-24522-AL17. Implementation of the modification is planned during the 2016 station

	Containment Outage. This project is in the PMC Preparation Phase.
References	NK21-CORR-00531-12282 NK21-CORR-00531-12123

CARD #	CA-0069
CARD Title	SIP-25: BA & BB New Neutronic Trips Feasibility Project
CARD Description	Per CNSC Correspondence: NK21-CORR-00531-12850 / NK29-CORR-00531-13310 - ACTION ITEM 1207-3320: SEMI ANNUAL UPDATE ON THE IMPLEMENTATION OF LINEAR RATE TRIPS
Applicable Units	Bruce A & Bruce B
Alert Group	DPTPROJB
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	New Neutronic Trips will have an immediate impact on maintaining and improving the design basis of SDSs.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved SDS capability, Column 1- Introduces new design features in support of improving effectiveness of SDS capability
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-026
GIO Title	BA & BB New Neutronic Trips
CARD(s) Associated with this GIO	CA-0069
Gap(s) Associated with this CARD	SF01-IIP-018-14
Additional Information	SIP-25 action and AI 1207-3320 are still in progress. Bruce Power is considering implementing the linear rate trips as physical design improvements in Bruce B, and is currently pursuing installation of Design Demonstration Units (DDUs) of the linear rate trip on the lead unit to demonstrate the feasibility of the design. With respect to the SDS1 DDUs, they were delivered and will be under Bruce Power Control and maintenance testing and calibration; the documents in support of installation are under development; and the Bruce Power Plant Design and Engineering team has incorporated, as part of the DDU, a new requirement that Class II power supply be available.

	With respect to the SDS2 DDUs, the vendor has sent a revised delivery schedule to Bruce Power. The installation schedule is contingent upon the final equipment delivery schedule from the vendors. The next update for this Action Item will be in December 2017. Note that the TCD is a milestone date. Upon completion of the milestone, the TCD will be updated with the implementation date. The IIP item will remain open until implementation has been confirmed.
References	NK21 -CORR-00531 -12850 / NK29-CORR-00531 -13310 NK21-CORR-00531-12491 / NK29-CORR-00531-12909AI 1207-3320 NK21-CORR-00531-11357 NK29-CORR-00531-11762 AI 1207-3320

CARD #	CA-0297
CARD Title	Implement SDS2 Neutron Overpower Protection Enhancements - Unit 3
CARD Description	The SDS2 NOP Enhancement project includes a number of improvements to SDS2 equipment and performance, the primary one being the addition of 24 vertical NOP detectors to enhance SDS2 coverage of a number of perturbed flux shapes and expand SDS2 NOP coverage similar to that provided by SDS1. Safety analysis in support of SDS2 NOP enhancement has been performed and submitted to the CNSC, and CNSC approval has been granted for the design change. DCP 3447
Applicable Units	Unit 3
Alert Group	DIVMCR
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Implementing SDS2 Neutron Overpower Protection Enhancements will have an immediate impact on maintaining and improving the design basis of SDS2.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved SDS2 capability, Column 1- Introduces new design features in support of improving effectiveness of SDS2 capability.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	30-Jun-26
GIO #	GIO-090
GIO Title	SDS2 Enhancements
CARD(s) Associated with this GIO	CA-0297, CA-0378
Gap(s) Associated with this CARD	SF01-SUP-001-16
Additional Information	AMOT-0366
References	

CARD #	CA-0378
CARD Title	Implement SDS2 Neutron Overpower Protection Enhancements - Unit 4
CARD Description	The SDS2 NOP Enhancement project includes a number of improvements to SDS2 equipment and performance, the primary one being the addition of 24 vertical NOP detectors to enhance SDS2 coverage of a number of perturbed flux shapes and expand SDS2 NOP coverage similar to that provided by SDS1. Safety analysis in support of SDS2 NOP enhancement has been performed and submitted to the CNSC, and CNSC approval has been granted for the design change. DCP 3447
Applicable Units	Unit 4
Alert Group	DIVMCR
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Implementing SDS2 Neutron Overpower Protection Enhancements will have an immediate impact on maintaining and improving the design basis of SDS2.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved SDS2 capability, Column 1- Introduces new design features in support of improving effectiveness of SDS2 capability.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	31-Dec-27
GIO #	GIO-090
GIO Title	SDS2 Enhancements
CARD(s) Associated with this GIO	CA-0297, CA-0378
Gap(s) Associated with this CARD	SF01-SUP-049-16
Additional Information	AMOT-0366
References	

CARD #	CA-0299
CARD Title	BA ASB Fire Protection Upgrades
CARD Description	This initiative was implemented under Project # 32100 - Bruce A ASB Fire Protection Upgrades.
Applicable Units	Bruce A
Alert Group	DIVOPA
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	22-Dec-17
GIO #	GIO-091
GIO Title	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0299, CA-0300, CA-0301, CA-0302, CA-0303, CA-0304
Gap(s) Associated with this CARD	SF01-SUP-002-16
Additional Information	Completed per project 32100 & DCP0002949. Residual actions are being tracked by AR 28589533 (DM revisions & battery calculations, TCD = 18Nov2018). This will be removed as an active CARD in the next progress report.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659

CARD #	CA-0300
CARD Title	Unit 1 and 2 Fire Upgrades (Restart - Project #38730)
CARD Description	This initiative is implemented under Project # 38730 - BA U1&2 Fire Upgrades. 1) Complete the remaining work associated with Restart DCNs to install the turbine sprinkler Fire Alarm Boxes and associated power supplies, thereby removing fire protection related devices from the elevator room. 2) Install a direct connection to the IMS office fire protections. 3) Complete the work under DCN39202 for installation of RTDs downstream of the charcoal filters in Unit 0.
Applicable Units	Units 1 and 2
Alert Group	DPTEP
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	30-Jun-20
GIO #	GIO-091
GIO Title	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0299, CA-0300, CA-0301, CA-0302, CA-0303, CA-0304
Gap(s) Associated with this CARD	SF01-SUP-003-16
Additional Information	This project is in the PMC Definition Phase. DCNs 15373, 15375, 66370 and 66372 were completed in Q1 2016. Bruce Power will execute the work under WO 1726550 and 1726548. The Work Orders are currently scheduled for Q4 2018. Upon completion of the design closeout, DCP 3267 will be closed.
References	NK21-CORR-00531-13031

CARD #	CA-0301
CARD Title	BA Standby Generator Building Fire Protection Upgrade
CARD Description	This initiative is implemented under Project # 31723 - Bruce A Standby Generator Building Fire Protection Upgrade. Provide fire detection to address the separation between the SG buildings and the Fuel oil pumphouse or/ other SG buildings being less than the allowable separation distance of 6ft per NBC.
Applicable Units	Bruce A
Alert Group	DIVOPA
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	17-Dec-21
GIO #	GIO-091
GIO Title	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0299, CA-0300, CA-0301, CA-0302, CA-0303, CA-0304
Gap(s) Associated with this CARD	SF01-SUP-004-16
Additional Information	This project is being tracked under RegM AR 28460673.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659

CARD #	CA-0302
CARD Title	Bruce A Fire Barriers Upgrades (Cable Wraps)
CARD Description	This initiative is implemented under Project 39370 - Fire Protection: Bruce A Fire Protective Cable. Implement fire barrier improvements (cable tray wraps).
Applicable Units	Bruce A
Alert Group	DIVOPA
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-091
GIO Title	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0299, CA-0300, CA-0301, CA-0302, CA-0303, CA-0304
Gap(s) Associated with this CARD	SF01-SUP-005-16
Additional Information	This project is being tracked under RegM AR 28460667.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659

CARD #	CA-0303
CARD Title	Bruce A Very Early Smoke Detection Apparatus (VESDA) Upgrade
CARD Description	This initiative is implemented under Project # 39318 - BA Fire Protection: VESDA Upgrade MCR/CER. - Gap Analysis required to identify changes required; - Implementation of the changes required
Applicable Units	Bruce A
Alert Group	DIVOPA
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-091
GIO Title	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0299, CA-0300, CA-0301, CA-0302, CA-0303, CA-0304
Gap(s) Associated with this CARD	SF01-SUP-006-16
Additional Information	This project is in the PMC Development Phase.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955 NK21-CORR-00531-11324/ NK29-CORR-00531-11729

CARD #	CA-0304
CARD Title	Unit 1 and 2 Fire Upgrades (SCA VESDA & Turbine Sprinkler System alarm detection and notification interface)
CARD Description	This initiative is implemented under Project # 39319. BA Fire Protection: Unit 1 and 2 Fire Upgrades (Restart) - DCP 3270
Applicable Units	Units 1 and 2
Alert Group	DPTEP
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-091
GIO Title	Bruce A Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0299, CA-0300, CA-0301, CA-0302, CA-0303, CA-0304
Gap(s) Associated with this CARD	SF01-SUP-007-16
Additional Information	This project is in the PMC Development Phase. Racer ID 2511 has been prepared for the 2017 business plan. Bruce Power will ensure that the design is updated to show removal of the fire protection devices and physically remove them in the field.
References	NK21-CORR-00531-13031

CARD #	CA-0306
CARD Title	BB U0 Fuel Storage Area Sprinkler Upgrades
CARD Description	This initiative is implemented under Project # 37465 - BB U0 Fuel Storage Area Sprinkler Upgrades. Upgrade existing system to increase capacity and allow additional new fuel storage.
Applicable Units	Unit 0B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-009-16
Additional Information	This project is in the PMC Preparation Phase.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0307
CARD Title	Bruce B Fireworks Terminal Replacement
CARD Description	This initiative is implemented under Project # 38745 - BB Fireworks Terminal Replacement. Replace obsolete fireworks terminals.
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	20-Dec-19
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-010-16
Additional Information	This initiative is implemented under Project # 38745. This project is in the PMC Development Phase. Replace obsolete fireworks terminals.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0308
CARD Title	Bruce B Firewater Pipe Replacement
CARD Description	This initiative is implemented under Project # 38743 - BB Firewater Pipe Replacement. Replace Old Water Treatment Plant tunnel pipe.
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-011-16
Additional Information	This project is in the PMC Development Phase.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0309
CARD Title	Bruce B Fire Detection Upgrade
CARD Description	This initiative is implemented under Project # (Racer 2111) - Bruce B Fire Detection Upgrade. Addition of automatic detection at various locations in BB as an outcome of the FSSA and FHA.
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	17-Dec-21
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-012-16
Additional Information	This project is being tracked under RegM AR 28460729
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0310
CARD Title	Bruce B Very Early Smoke Detection Apparatus (VESDA) Upgrade
CARD Description	This initiative is implemented under Project # (Racer 2110) - Bruce B VESDA Upgrade. Upgrade VESDA in 13 rooms in Bruce B.
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-013-16
Additional Information	This project is being tracked under RegM AR 28531785.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955 NK21-CORR-00531-11324/ NK29-CORR-00531- 11729

CARD #	CA-0311
CARD Title	Bruce B Fire Barriers (Cable Wrap) upgrades
CARD Description	This initiative is implemented under Project # 39346 - Bruce B Fire Barriers (Cable Wraps) Upgrades. Implement fire barrier improvements (cable tray wraps).
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-014-16
Additional Information	
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0312
CARD Title	Bruce B Standby Generator Building Fire Protection Upgrade
CARD Description	This initiative is implemented under Project # 31711 - BB Standby Generator Building Fire Protection Upgrade.
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	17-Dec-21
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-015-16
Additional Information	This project is in the PMC Development Phase.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0313
CARD Title	BB EPG / EWPS Building Fire Protection Upgrade
CARD Description	This initiative is implemented under Project # 31712. - BB EPG / EWPS Building Fire Protection Upgrade.
Applicable Units	Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	17-Dec-21
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-016-16
Additional Information	This project is in the PMC Development Phase.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0315
CARD Title	Unit 8 Fire Upgrades
CARD Description	This initiative is implemented under Project #(Racer 2508) - Unit 8 Fire Upgrades - DCP 3328 Installation of auto dialer in the U8 IMS Modular Office.
Applicable Units	Unit 8
Alert Group	DPTEP
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Underway
Target Completion Date	18-Dec-20
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-018-16
Additional Information	This project is in the PMC Initiation Phase.
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659

CARD #	CA-0316
CARD Title	Air Foam System Replacement
CARD Description	This initiative is implemented under Project # (Racer 2116) - Air Foam System Replacement.
Applicable Units	Bruce A & Bruce B
Alert Group	DIVOPB
Functional Area	DPTPRG
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Fire Protection upgrades will have an immediate impact on enhanced fire safety and alignment with CSA-N293-07.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improving design features for improved fire protection, Column 1- Introduces new design features in support of improving effectiveness of fire protection and safety.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	18-Dec-20
GIO #	GIO-092
GIO Title	Bruce B Fire Protection Upgrades to Align with CSA-N293-07
CARD(s) Associated with this GIO	CA-0306, CA-0307, CA-0308, CA-0309, CA-0310, CA-0311, CA-0312, CA-0313, CA-0315, CA-0316
Gap(s) Associated with this CARD	SF01-SUP-019-16
Additional Information	
References	NK21-CORR-00531-13173 / NK29-CORR-00531-13659 NK21-CORR-00531-11574 / NK29-CORR-00531-11955

CARD #	CA-0349
CARD Title	Implementation - Legacy Registration Project DCN/DCPs- Bruce A
CARD Description	Implement field modifications per the Bruce A Legacy Registration Project DCNs/DCPs
Applicable Units	Bruce A
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design through modifications and its configuration management, Column 2- Completion of legacy registration augments effectiveness of the current practices.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-097
GIO Title	Bruce A Legacy Registration- Implementation Projects
CARD(s) Associated with this GIO	CA-0298, CA-0349
Gap(s) Associated with this CARD	SF01-SUP-021-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note that the TCD is a milestone date. Upon completion of the milestone, the TCD will be updated with the implementation date. The IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0351
CARD Title	Implementation - Legacy Registration Project DCN/DCPs- Bruce B
CARD Description	Implement field modifications per the Bruce B Legacy Registration Project DCNs/DCPs
Applicable Units	Bruce B
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	5
Impact Attribute Rationale	Impact Evaluation Matrix: Row 1- Relates to improvement of the design through modifications and its configuration management, Column 2- Completion of legacy registration augments effectiveness of the current practices.
Time-Impact Utility Score	1.00000
Final Score	0.00080
CARD Priority	23
CARD Status	Not Started
Target Completion Date	21-Dec-18
GIO #	GIO-098
GIO Title	Bruce B Legacy Registration- Implementation Projects
CARD(s) Associated with this GIO	CA-0351
Gap(s) Associated with this CARD	SF01-SUP-023-16
Additional Information	This will be implemented by PSE-OMA-81406 project. Note: The TCD is a milestone date; upon completion of milestone, TCD will be updated with implementation date. IIP item will remain open until implementation has been confirmed.
References	

CARD #	CA-0066
CARD Title	SIP-22: Enhanced Periodic Safety Review to Support Asset Management
CARD Description	Bruce Power is utilizing an Asset Management approach to ensure safe plant operations throughout its life cycle. A Periodic Safety Review (PSR) process will be used to demonstrate and improve safety throughout the plant operating life. The PSR will be further enhanced by a safety basis process and Composite Safety Profile (CSP) to provide an integrated measure of safety and predicted change in safety of the plant on year over year basis for a defined period. The CSP will combine probabilistic and deterministic safety risk issues into a consistent measure of overall plant safety. The safety basis process and CSP ranking of safety issues will be used to identify the significant risks areas in order to help optimized plant safety improvement and the IIP. Implementation of the IIP will ensure safety is maintained and improved in a cost effective manner.
Applicable Units	Bruce A & Bruce B
Alert Group	DPTNSAS
Functional Area	DIVDMES
Value Tree Tier 3 Objective	5.4.1 - Enhanced confidence in management system structure, and processes and supporting infrastructure
Tier 3 Weight	0.00221
Time Attribute Score	5
Time Attribute Rationale	Enhanced Periodic Safety Review to Support Asset Management will take 1-2 years to have an impact in Enhanced confidence in the governance and processes
Impact Attribute Score	2
Impact Attribute Rationale	Impact Evaluation Matrix: Row 4b- Relates to improving the managed system, Column 3- Improves the current practices in place.
Time-Impact Utility Score	0.26259
Final Score	0.00058
CARD Priority	24
CARD Status	Underway
Target Completion Date	22-Dec-17
GIO #	GIO-024
GIO Title	Enhanced Periodic Safety Review to Support Asset Management
CARD(s) Associated with this GIO	CA-0066
Gap(s) Associated with this CARD	SF10-IIP-019-14
Additional Information	SIP-22 action is in progress. Procedure BP-PROC-01024 has been issued which establishes and describes the requirements and processes for the conduct of Periodic Safety Reviews (PSRs), for the purpose of plant life cycle management, in accordance with the requirements licence condition 15.2 of the Bruce A and Bruce B Power Reactor Operating Licence and Section 15.2 of the Licence Conditions Handbook (LCH); this includes:

	Progress continues in the development of the safety basis process. NK21 - CORR-00531 -12269 provides the timeline for the expected submissions for both Bruce A and Bruce B.
References	NK21-CORR-00531-12269 NK21-CORR-00531-10576 NK29-CORR-00531-10975

CARD #	CA-0298
CARD Title	Documentation - Legacy Registration Project DCN/DCPs- Bruce A
CARD Description	Update design documentation per the Bruce A Legacy Registration Project DCNs/DCPs.
Applicable Units	Bruce A
Alert Group	DPTMCD
Functional Area	DIVDMES
Value Tree Tier 3 Objective	1.3.1 - Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards
Tier 3 Weight	0.00083
Time Attribute Score	5
Time Attribute Rationale	Once resolved will have an immediate effect on the objective.
Impact Attribute Score	3
Impact Attribute Rationale	Impact Evaluation Matrix: Row 3- Relates to improvement of the design documentation and its configuration management, Column 2- Completion of legacy registration augments effectiveness of the current practices.
Time-Impact Utility Score	0.46118
Final Score	0.00037
CARD Priority	25
CARD Status	Underway
Target Completion Date	31-May-17
GIO #	GIO-097
GIO Title	Bruce A Legacy Registration- Implementation Projects
CARD(s) Associated with this GIO	CA-0298, CA-0349
Gap(s) Associated with this CARD	SF01-SUP-020-16
Additional Information	This has been completed, and will be removed as an active CARD in the next progress report.
References	NK21-CORR-00531-11941 NK21-CORR-00531-09328 NK21-CORR-00531-08728 NK21-CORR-00531-08217 NK21-CORR-00531-05602