FORM-14159 R000*

Periodic Safety Review - Final Document Review Traveler

Bruce Power

Bruce Power Document #:	Revision:	Information Classification	Usage Classification
NK29-SFR-09701-00004	R000	Internal Use Only	Information

Bruce Power Document Title:
Safety Factor 4 – Ageing

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

Supplier's Name:	Supplier Document #:	Revision:
CANDESCO	K-421231-00204	R00

Supplier Document Title:
Safety Factor 4 – Ageing

Accepted for use at Bruce Power by:	**Signature:**	**Date**
Name: Gary Newman Title: Chief Engineer & Sr. Vice President, Engineering	G Newman	30SEP2016

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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Reviewed By:

Name	Title	Department	Signature	Date
David Cho	Sr. Technical Specialist	Nuclear Safety Analysis and Support	electronic acceptance	25Aug2016
Andrew Glover	Sr. Technical Specialist	Nuclear Safety Analysis and Support	electronic acceptance	16Aug2016
Jason Goldberg (RegC AR 28563014.07)	Department Manager	Nuclear Safety Analysis and Support		
Andrew Maksymyk (RegC AR 28563014.02)	Department Manager	Reactor Safety Support		

Recommended for Use By:

Name	Title	Department	Signature	Date
Peter Purdy (RegC AR 28563014)	Division Manager	Reactor Safety Engineering		Sep. 29/16

NOTE: Per meeting held with VP, NORA on September 28, 2016, concerns with wording of some of the macro gaps will be addressed with the addition of context-setting text in the GAR and IIP documents. P.P. Sep. 29/16

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Reviewed By:

Name	Title	Department	Signature	Date
David Cho	Sr. Technical Specialist	Nuclear Safety Analysis and Support	electronic acceptance	25Aug2016
Andrew Glover	Sr. Technical Specialist	Nuclear Safety Analysis and Support	electronic acceptance	16Aug2016
Jason Goldberg (RegC AR 28563014.07)	Department Manager	Nuclear Safety Analysis and Support		
Andrew Maksymyk (RegC AR 28563014.02)	Department Manager	Reactor Safety Support	A. Maksymyk	28 Sep 2016

Recommended for Use By:

Name	Title	Department	Signature	Date
Peter Purdy (RegC AR 28563014)	Division Manager	Reactor Safety Engineering		

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Reviewed By:

Name	Title	Department	Signature	Date
David Cho	Sr. Technical Specialist	Nuclear Safety Analysis and Support	electronic acceptance	25Aug2016
Andrew Glover	Sr. Technical Specialist	Nuclear Safety Analysis and Support	electronic acceptance	16Aug2016
Jason Goldberg (RegC AR 28563014.07)	Department Manager	Nuclear Safety Analysis and Support	[signature]	29SEP2016
Andrew Maksymyk (RegC AR 28563014.02)	Department Manager	Reactor Safety Support		

Recommended for Use By:

Name	Title	Department	Signature	Date
Peter Purdy (RegC AR 28563014)	Division Manager	Reactor Safety Engineering		

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Reviewed By:

Name	**Title**	**Department**	**Signature**	**Date**
Martin Baumann	Shift Supervisor In Training	Bruce B Operations	electronic acceptance	28Aug2016
Colin Moranno (RegC AR 28562999.03)	Division Manager	Bruce B Operations	C. Moranno	23Sep2016

Recommended for Use By:

Name	**Title**	**Department**	**Signature**	**Date**
Paul Clark (RegC AR 28562999)	Plant Manager	Bruce B Operations	P N Clark	23Sep16

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Reviewed By:

Name	Title	Department	Signature	Date
Steven Ostrowski	Department. Manager	Component Design & Design Programs	electronic acceptance	23Aug2016
Dennis Kazimer (RegC AR 28563002.04)	Department Manager (delegate)	Component Design & Design Programs	D Kazimer	23 Sept 2016

Recommended for Use By:

Name	Title	Department	Signature	Date
Gord Kozak (RegC AR 28563002)	Division Manager	Engineering Support		29 SEPT 2016

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Reviewed By:

Name	Title	Department	Signature	Date
Ian Cruchley	Section Manager	Major Components Engineering	electronic acceptance	23Aug2016
Stephen Miller (RegC AR 28563003.03)	Department Manager	Equipment Reliability Integration	Stephen Miller	21 SEP 2016

Recommended for Use By:

Name	Title	Department	Signature	Date
Kevin Pickles (RegC AR 28563003)	Division Manager	Station Engineering		22SEP2016

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Reviewed By:

Name	Title	Department	Signature	Date

Recommended for Use By:

Name	Title	Department	Signature	Date

candesco
Division of Kinectrics Inc.

NK29-SFR-09701-00004

Title: Safety Factor 4 - Ageing

File: K-421231-00204-R00

A Report Submitted to Bruce Power

September 20, 2016

Issue					
Issue R00D0	**Reason for Issue:** For harmonization				
	Author: C. Oberholzer	Verifier:	Reviewer:	Approver:	Date: April 25, 2016
Issue R00D1	**Reason for Issue:** For internal review				
	Author: C. Oberholzer	Verifier:	Reviewer: L. Watt G. Archinoff	Approver:	Date: April 27, 2016
Issue R00D2	**Reason for Issue:** For Bruce Power review				
	Author: C. Oberholzer	Verifier: G. Buckley	Reviewer: L. Watt G. Archinoff	Approver:	Date: May 19, 2016
Issue R00D3	**Reason for Issue:** Addresses Bruce Power review comments and internal verification comments.				
	Author: C. Oberholzer	Verifier: G. Buckley	Reviewer: L. Watt G. Archinoff	Approver:	Date: August 24, 2016

Issue	**Reason for Issue:**				
R00	For use				
	Author: C. Oberholzer	Verifier: G. Buckley	Reviewer: L. Watt G. Archinoff	Approver: L. Watt	Date: Sept 20, 2016
Document Classification: Report			Security Classification: Client Proprietary		

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Acronyms and Abbreviations

ACT	Asset Challenge Team
AHJ	Authority Having Jurisdiction
ALPO/ALP&O	Asset Life Projections and Options
AMP	Ageing Management Program
AR	Action Request
BP	Bruce Power
Bruce 1&2	Bruce Units 1 and 2
CANDU	Canada Deuterium Uranium
CBM	Condition Based Maintenance
CCP	Critical Channel Power
CERI	Continuing Equipment Reliability Improvement
CFAM	Corporate Functional Area Manager
CHIP	Component Health Improvement Plan
CHR	Component Health Report
CNSC	Canadian Nuclear Safety Commission
CPMP	Component Performance Monitoring Plan
Crit-Cat	Criticality Category
CSA	Canadian Standards Association
CSI	CANDU Safety Issue
CSM	Catches, Saves and Misses
D	Deuterium
DEM	Duty Engineering Manager
DHC	Delayed Hydride Cracking
EACE	Equipment Apparent Cause Evaluation
EPRI	Electric Power Research Institute
EQ	Environmental Qualification
ER	Equipment Reliability
ERCI	Equipment Root Cause Investigation
FAC	Flow Accelerated Corrosion

FASA	Focus Area Self-Assessment
FCAMP	Fuel Channel Ageing Management Plan
FCCA	Fuel Channel Condition Assessment
FCLCMP	Fuel Channel Life Cycle Management Plan
FCLMP	Fuel Channel Life Management Program
FMEA	Failure Modes and Effects Analysis
GE	General Electric
HTS	Heat Transport System
I&C	Instrumentation and Control
IAEA	International Atomic Energy Agency
IGALL	International Generic Ageing Lessons Learned
INPO	Institute of Nuclear Power Operations
ISR	Integrated Safety Review
ITP	Inspection and Test Plans
JIT	Just-in-Time
LBLOCA	Large Break Loss-of-Coolant Accident
LCH	Licence Condition Handbook
LCMP	Life Cycle Management Plan
LOCA	Loss-of-Coolant Accident
LOF	Loss of Flow
LTEP	Long Term Energy Plan
LTO	Long Term Operation
MCR	Major Component Replacement
MEL	Master Equipment List
NOP	Neutron Overpower
NPP	Nuclear Power Plant
NSA	Nuclear Safety Assessment
NSCA	Nuclear Safety and Control Act
NUOG	Nuclear Utility Obsolescence Group
O&M	Operations and Maintenance
OBSE	Obsolescence (action plan type)
OFI	Opportunities for Improvement

OIRD	Obsolescence Items Replacement Database
OLCs	Operational Limits and Conditions
OMS	Outage and Maintenance Services
OOC	Obsolescence Oversight Committee
OPC	Obsolescence Process Coordinator
OPEX	Operating Experience
OPP	Operating Policies and Principles
OSART	Operational Safety Review Team
OSR	Operational Safety Requirements
OWSC	Obsolescence Working Solutions Committee
PdM	Predictive Maintenance
PHT	Primary Heat Transport
PIE	Postulated Initiating Event
PIP	Periodic Inspection Program
PM	Preventive Maintenance
PMEL	Performance Monitoring Equipment List
PMIDRQ	Preventative Maintenance Identification Requirement
POMS	Proactive Obsolescence Management System
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PT-CT	Pressure Tube-Calandria Tube
R&D	Research & Development
RCE	Responsible Component Engineer
RCM	Risk Control Measures
RIDM	Risk-Informed Decision Making
RPE	Responsible Program Engineer
RS	Reactor Safety
RSE	Responsible System Engineer
SBLOCA	Small Break Loss-of-Coolant Accident
SBR	Safety Basis Report
SCR	Station Condition Record

SEBOB	Station Engineering Basis Oversight Board
SFR	Safety Factor Report
SG	Steam Generator
SHIP	System Health Improvement Plan
SHR	System Health Reports
SIS	Systems Important to Safety
SOE	Safe Operating Envelope
SPHC	Station Plant Health Committee
SPMP	System Performance Monitoring Plan
SPV	Single Point of Vulnerability
SSC	Structures, Systems, and Components
SST	Safety-Related System Testing
TBA	Technical Basis Assessments
TOE	Technical Operability Evaluation
TOQ	Task Order Quotation
WANO	World Association of Nuclear Operators
WO	Work Order

1. Objective and Description

Bruce Power (BP), as an essential part of its operating strategy, is planning to continue operation of Bruce B as part of its contribution to the Long Term Energy Plan (LTEP) (http://www.energy.gov.on.ca/en/ltep/). Bruce Power has developed integrated plant life management plans in support of operation to 247,000 Equivalent Full Power Hours in accordance with the Bruce Power Reactor Operating Licence (PROL) [1] and Licence Conditions Handbook (LCH) [2]. A more intensive Asset Management program is under development, which includes a Major Component Replacement (MCR) approach to replacing pressure tubes, feeders and steam generators, so that the units are maintained in a fit for service state over their lifetime. However, due to the unusually long outage and de-fuelled state during pressure tube replacement, there is an opportunity to conduct other work, and some component replacements that could not be done reasonably in a regular maintenance outage will be scheduled concurrently with MCR. In accordance with Licence Condition 15.2 of the PROL [1], Bruce Power is required to inform the Canadian Nuclear Safety Commission (CNSC) of any plan to refurbish a reactor or replace a major component at the nuclear facilities, and Bruce Power shall:

(i) Prepare and conduct a periodic safety review;

(ii) Implement and maintain a return-to-service plan; and

(iii) Provide periodic updates on progress and proposed changes.

The fifteen reports prepared as part of the Periodic Safety Review (PSR), including this Safety Factor Report (SFR), are intended to satisfy Licence Condition 15.2 (i) as a comprehensive evaluation of the design, condition and operation of the nuclear power plant (NPP). In accordance with Regulatory Document REGDOC-2.3.3 [3], a PSR is an effective way to obtain an overall view of actual plant safety and the quality of safety documentation and determine reasonable and practicable improvements to ensure safety until the next PSR.

Bruce Power has well-established PSR requirements and processes for the conduct of a PSR for the purpose of life-cycle management, which are documented in the procedure Periodic Safety Reviews [4]. This procedure, in combination with the Bruce B Periodic Safety Review Basis Document [5], governs the conduct of the PSR and facilitates its regulatory review to ensure that Bruce Power and the CNSC have the same expectations for scope, methodology and outcome of the PSR.

This PSR supersedes the Bruce B portion of the interim PSR that was conducted in support of the ongoing operation of the Bruce A and Bruce B units until 2019 [6]. Per REGDOC-2.3.3 [3], subsequent PSRs will focus on changes in requirements, facility conditions, operating experience (OPEX) and new information rather than repeating activities of previous reviews.

1.1. Objective

The overall objectives of the Bruce B PSR are to conduct a review of Bruce B against modern codes and standards and international safety expectations, and to provide input to a practicable set of improvements to be conducted during the MCR in Units 5 to 8, and during asset management activities to support ongoing operation of all four units, as well as U0B, that will enhance safety to support long term operation. It will cover a 10-year period, since there is an expectation that a PSR will be performed on approximately a 10-year cycle, given that all units are expected to be operated well into the future.

The specific objective of the review of this Safety Factor is to determine whether ageing[1] aspects affecting Structures, Systems and Components (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.

1.2. Description

The review is conducted in accordance with the Bruce B PSR Basis Document [5], 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

[1] In this Safety Factor Report, "ageing" and "aging" are used interchangeably. Bruce Power documents generally use "aging" while the IAEA's SSG-25 [48] uses "ageing".

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.

As required by the PSR Basis Document, preparation of this Safety Factor Report included an assessment of the review tasks to determine if modifications were appropriate. Any changes to the review tasks described in this section are documented and justified in Section 5.

2. Methodology of Review

As discussed in the Bruce B PSR Basis Document [5], the methodology for a PSR should include making use of safety reviews that have already been performed for other reasons. Accordingly, the Bruce B PSR makes use of previous reviews that were conducted for the following purposes:

- Return to service of Bruce Units 3 and 4 (circa 2001) [7];
- Life extension of Bruce Units 1 and 2 (Bruce 1&2) (circa 2006) [8] [9] [10];
- Proposed refurbishments of Bruce Units 3 and 4 (circa 2008) [11] [12] [13] [14] [15];
- Safety Basis Report (SBR) and PSR for Bruce Units 1 to 8 (2013) [6]; and
- Bruce A Integrated Safety Review (ISR) to enhance safety and support long term operation (2015) [16] [17].

These reviews covered many, if not all, of the same Safety Factors that are reviewed in the current PSR. A full chronology of Bruce Power safety reviews up to 2013 is provided in Appendix F of [18].

The Bruce B PSR Safety Factor review process comprises the following steps:

1. **Interpret and confirm review tasks:** As a first step in the Safety Factor review, the Safety Factor Report author(s) confirm the review tasks identified in the PSR Basis Document [5] and repeated in Section 1.2 to ensure a common understanding of the intent and scope of each task. In some cases, this may lead to elaboration of the review tasks to ensure that the focus is precise and specific. Any changes to the review tasks are identified in Section 5 of the Safety Factor Report (SFR) and a rationale provided.
2. **Confirm the codes and standards to be considered for assessment:** The Safety Factor Report author(s) validates the list of codes and standards presented in the PSR Basis Document against the defined review tasks to ensure that the assessment of each standard will yield sufficient information to complete the review tasks. Additional codes and standards are added if deemed necessary. If no standard can be found that covers the review task, the assessor may have to identify criteria on which the assessment of the review task will be based. The final list of codes and standards considered for this Safety Factor is provided in Section 7.
3. **Determine the type and scope of assessment to be performed:** This step involves the assessor confirming that the assessment type identified in Appendix C of the Bruce B PSR

Basis Document [5] for each of the codes, standards and guidance documents selected for this factor is appropriate based on the guidance provided. The PSR Basis Document provides an initial assignment for the assessment type, selecting one of the following review types:

- Programmatic Clause-by-Clause Assessments;
- Plant Clause-by-Clause Assessments;
- High-Level Programmatic Assessments;
- High-Level Plant Assessments;
- Code-to-Code Assessments; or
- Confirm Validity of Previous Assessment.

The final assessment types are identified in Section 3, along with the rationale for any changes relative to the assignment types listed in the PSR Basis Document.

4. **Perform gap assessment against codes and standards:** This step comprises the actual assessment of the Bruce Power programs and the Bruce B plant against the identified codes and standards. In general, this involves determining from available design or programmatic documentation whether the plant or program meet the provisions of the specific clause of the standard or of some other criterion, such as a summary of related clauses. Each individual deviation from the provisions of codes and standards is referred to as a Safety Factor “micro-gap”. The assessments, performed in Appendix A and Appendix B, include the assessor’s arguments conveying reasons why the clause is considered to be met or not met, while citing appropriate references that support this contention.
5. **Assess alignment with the provisions of the review tasks:** The results of the assessment against codes and standards are interpreted in the context of the review tasks of the Safety Factor. To this end, each assessment, whether clause-by-clause, high-level or code-to-code, is assigned to one or more of the review tasks (Section 5). Assessment against the provision of the review task involves formulating a summary assessment of the degree to which the plant or program meets the objective and provisions of the particular review task. This assessment may involve consolidation and interpretation of the various compliance assessments to arrive at a single compliance indicator for the objective of the review task as a whole. The results of this step are documented in Section 5 of each SFR.
6. **Perform program assessments:** The most pertinent self-assessments, audits and regulatory evaluations are assessed, and performance indicators relevant to the Safety Factor identified. The former illustrates that Bruce Power has a comprehensive process of reviewing compliance with Bruce Power processes, identifying gaps, committing to corrective actions, and following up to confirm completion and effectiveness of these actions. The latter demonstrates that there is a metric by which Bruce Power assesses the effectiveness of the programs relevant to the Safety Factor in Section 7. Taken as a whole, these demonstrate that the processes associated with this Safety Factor are implemented effectively (individual findings notwithstanding). Thus, program effectiveness, if not demonstrated explicitly in the review task assessments in Step 5, can be inferred if Step 5

shows that Bruce Power processes meet the Safety Factor requirements and if this step shows there are ongoing processes to ensure compliance with Bruce Power processes.

7. **Identification of findings:** This step involves the consolidation of the findings of the assessment against codes and standards and the results of executing the review tasks into a number of definitive statements regarding positive and negative findings of the assessment of the Safety Factor. Positive findings or strengths are only identified if there is clear evidence that the Bruce B plant or programs exceed compliance with the provision of codes and standards or review task objectives. Each individual negative finding or deviation is designated as a Safety Factor micro-gap for tracking purposes. Identical or similar micro-gaps are consolidated into comprehensive statements that describe the deviation known as Safety Factor macro-gaps, which are listed in Section 8 of the Safety Factor Reports, as applicable.

3. Applicable Codes and Standards

This section lists the applicable regulatory requirements, codes and standards considered in the review of this Safety Factor. Table C-1 of the Bruce B PSR Basis Document [5] identifies the codes, standards and guides that are relevant to this PSR. Modern revisions of some codes and standards listed in Table C-1 of the PSR Basis Document [5] have been identified in the licence renewal application and supplementary submissions for the current PROL [19] [20] [21]. Codes, standards and guides issued after the freeze date of December 31, 2015 were not considered in the review [5].

3.1. Acts and Regulations

The *Nuclear Safety and Control Act* (NSCA) [22] establishes the Canadian Nuclear Safety Commission and its authority to regulate nuclear activities in Canada. Bruce Power has a process to ensure compliance with the NSCA [22] and its Regulations. Therefore, the NSCA and Regulations were not considered further in this review.

3.2. Power Reactor Operating Licence

The list of codes and standards related to ageing that are referenced in the PROL [1] and LCH [2], and noted in Table C-1 of the Bruce B PSR Basis Document [5], are identified in Table 1. The edition dates referenced in the third column of the table are the modern versions used for comparison.

The licence conditions in the PROL [1] and LCH [2] that prescribe adherence to codes and standards of relevance to this safety factor are the following:

- Licence Condition 15.2 (i) on *Continued Operations* that requires the licensee to inform the Commission of any plan to refurbish a reactor or replace a major component at the nuclear facilities, and to prepare and conduct a periodic safety review;

- Licence Condition 3.3 on *Reporting Requirements* that requires the licensee to notify and report in accordance with REGDOC- 3.1.1 Reporting Requirements: Nuclear Power [23];
- Licence Condition 6.1 on *Fitness for Service* that requires the licensee to implement and maintain programs to ensure fitness for service of systems, structures and components, including an in-service inspection program for the safety significant balance of plant pressure retaining systems and components, and safety-related structures;
- Licence Condition 1.1 on *Management System Requirements* that requires the licensee to implement and maintain a management system; and,
- Licence Condition 5.3 on *Environmental Qualification Program* that requires the licensee to implement and maintain an environmental qualification program.

Collectively, these licence conditions invoke the codes and standards listed in Table 1 below.

Table 1: Codes, Standards, and Regulatory Documents Referenced in Bruce A and B PROL and LCH

Document Number	Document Title	Modern Version Used for PSR Comparison	Type of Review
CNSC REGDOC-2.3.3	Periodic Safety Reviews	[3]	NA
CNSC REGDOC-3.1.1	Reporting Requirements for Operating Nuclear Power Plants	[23]	NA
CNSC RD/GD-210	Maintenance Programs for Nuclear Power Plants	[24]	NA
CNSC RD/GD-98	Reliability Programs for Nuclear Power Plants	[25]	NA
CNSC REGDOC-2.6.3	Fitness for Service: Aging Management	[26]	NA
CSA-N285.4-09	Periodic Inspection of CANDU Nuclear Power Plant Components	CSA-N285.4-14 [27]	HL
CSA-N285.5-08	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	CSA-N285.5-13 [28]	CTC (HL)

Document Number	Document Title	Modern Version Used for PSR Comparison	Type of Review
CSA-N287.7-08	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	CSA-N287.7-08 (R2013) and Update 1 (Sep 2010) [29]	NA
CSA-N286-05	Management System Requirements for Nuclear Power Plants	CSA-N286-12 [30]	NA
CSA-N290.13-05 (R2010)	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	[31]	NA
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			

CNSC REGDOC-2.3.3: This PSR is being conducted in accordance with CNSC REGDOC-2.3.3 per Licence Condition 15.2 (i) [1], and associated compliance verification criteria [2]. Therefore, REGDOC-2.3.3 is not reviewed further in this document.

CNSC REGDOC-3.1.1: REGDOC-3.1.1 [23], Reporting Requirements for Nuclear Power Plants, is listed as Condition 3.3, Reporting Requirements, in the PROL [1] and sets reporting requirements for nuclear power plants including metrics related to pressure boundary degradation, plant reliability and preventive maintenance. This document has replaced S-99 [32] in the regulatory framework. The LCH [2] requires Bruce Power to transition to the new scheduled reporting, by June 30, 2015 for quarterly reports and by May 1, 2016 for annual reports but Bruce Power has already switched over to CNSC REGDOC-3.1.1 at the beginning of 2015, as committed in a letter submitted to the CNSC [21]. Line-by-line compliance with this regulatory document is verified on an ongoing basis to ensure compliance with the PROL, and therefore it was not assessed as part of this Safety Factor.

CNSC RD/GD-210: RD/GD-210 [24], Maintenance Programs for Nuclear Power Plants, is invoked by Condition 6.1, *Fitness for Service*, of the PROL [1] and outlines the requirements for a maintenance program. This document has replaced S-210 [33] in the regulatory framework. Requirements set out in RD/GD-210 [24] remain unchanged from those established in the eponymous S-210 [33], but adds information and guidance on how these requirements may be met. As a result of the Bruce 1&2 ISR, Bruce Power had committed to provide an assessment report of the maintenance program versus the intent of S-210 in 2008 [34]. The assessment for Bruce 1&2 was directly applicable to the Bruce 3&4 ISR and was not repeated at the time. Subsequently S-210 [33] has been included in the licence. A code-to-code comparison of RD/GD-210 [24] versus S-210 [33] with respect to ageing was performed in 2013 as part of the interim PSR and it was determined that RD/GD-210 [24] does not add to the requirements of S-210 [33]. Bruce Power is fully compliant with RD/GD-210 [24], as noted in Reference [21]. This is confirmed in the LCH [2] which requires Bruce Power to update the necessary documentation to change references from S-210 [33] to RD/GD-210 [24] in a systematic and timely manner as per their change management document by December 31, 2017. Since RD/GD-210 [24] is listed in the PROL [1], line-by-line compliance with this regulatory document is verified on an ongoing basis to ensure compliance with the PROL [1]. Therefore, as reflected in Table C-1 of the PSR Basis Document [5], assessment of RD/GD-210 [24] is not included in the review of this Safety Factor.

CNSC RD/GD-98: RD/GD-98 [25], Reliability Programs for Nuclear Power Plants, is invoked by Condition 6.1, *Fitness for Service*, of the PROL [1] and outlines the requirements for a reliability program for a nuclear power plant in Canada. This document has replaced the eponymous S-98 (Revision 1) [35] and now includes guidance on how reliability program requirements can be met. A review against S-98 [35] was completed for the Bruce 1 and 2 ISR and submitted to the CNSC and the program was established and implemented as required by the previous licence. RD/GD-98 [25] does not add to the requirements of S-98 [35] and continues to be a licence condition. The LCH [2] states that Bruce Power has prepared an implementation plan to transition to the requirements of RD/GD-98 [25] that includes the mapping between the existing RD/GD-98 [25] requirements and the Equipment Reliability program document. According to the LCH [2], Bruce Power was targeting completion of this mapping for December 2015. The latest version of the Equipment Reliability program document BP-PROG-11.01 [36] includes this mapping. Line-by-line compliance with this regulatory document is verified on an ongoing basis to ensure compliance with the PROL, and therefore, as reflected in Table C-1 of the PSR Basis Document [5], it was not assessed as part of the review of this Safety Factor.

CNSC REGDOC-2.6.3: REGDOC-2.6.3 [26], Aging Management, is invoked by Condition 6.1, *Fitness for Service*, of the PROL [1] and outlines the requirements related to aging management for SSCs of nuclear power plants in Canada. SSC-specific aging management programs (also, in some cases, referred to as Life Cycle Management Plans (LCMPs)), shall be implemented in accordance with the overall integrated aging management program framework. REGDOC-2.6.3 [26] replaced the eponymous RD-334 [37]. Bruce Power completed a gap assessment of Bruce Power governance against REGDOC-2.6.3 [26], and submitted a transition plan for implementation [38]. The gap assessment confirmed that the existing governance largely aligns with the requirements of REGDOC-2.6.3 [26], and identified some areas requiring clarification, for example, in the requirements for periodic reviews of aggregate effects of ageing, as well as governance considerations for ageing management during all phases of the lifecycle of the plant. The LCH [2] requires Bruce Power to achieve full compliance with REGDOC-2.6.3 [26] by June 30, 2016, with the exception of the LCMPs, when all milestones identified in their transition plan [38] are completed and the resulting revisions to governance and process documents are issued. Final implementation of all LCMPs, besides three major components (pressure tubes, feeders, steam generators), is targeted for completion by December 31, 2016. Therefore, as reflected in Table C-1 of the PSR Basis Document [5], no further assessment of CNSC REGDOC-2.6.3 [26] is necessary in the review of this Safety Factor.

CSA-N285.4-14: CSA-N285.4, Periodic Inspection of CANada Deuterium Uranium (CANDU) Nuclear Power Plant Components is invoked by Condition 6.1, *Fitness for Service*, of the PROL [1]. A new version of this standard was issued in 2009 [39] with an Update in 2011. The 2009 version with the 2011 Update is included in the PROL [1]. Since Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL, and since the 2009 version is subject to a transition plan, Table C-1 of the PSR Basis Document [5] indicates that compliance need not be assessed as part of this PSR. However, the latest version of this standard is N285.4-14 [27]. Therefore, a high level code-to-code comparison between the 2014 and 2009 versions was conducted and the results are presented in Appendix A (A.1).

CSA-N285.5-13: CSA-N285.5-08, Periodic Inspection of CANDU Nuclear Power Plant Containment Components is invoked under Condition 6.1, *Fitness for Service*, of the PROL [1]. Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance. However, the latest version of this standard is N285.5-13 [28], which supersedes that of N285.5-08. As a consequence Table C-1 of the PSR Basis Document [5] indicates that a high level code-to-code comparison between the 2013 and 2008 versions was to be conducted. The results of this assessment are presented in Appendix A (A.2).

CSA-N287.7-08: CSA-N287.7-08 [29], In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, is invoked under Condition 6.1, *Fitness for Service*, of the PROL [1]. Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance. There is not a newer version of this standard and its content was reaffirmed in 2013. Therefore, as reflected in Table C-1 of the PSR Basis Document [5], no further assessment of N287.7-08 [29] is necessary in the review of this Safety Factor.

CSA N286-12: CSA N286-05 is noted in the PROL (Licence Condition 1.1 [1]). Per the LCH [2], an implementation strategy for the 2012 version is in progress to be submitted to the CNSC by the end of January 2016. CNSC staff have stated that in their view the CSA N286-12 version of CSA N286 “does not represent a fundamental change to the current Bruce Power Management System” and have acknowledged that “the new requirements in CSA N286-12 are already addressed in Bruce Power's program and procedure documentation” [40].

Bruce Power had agreed to perform a gap analysis and to prepare a detailed transition plan, and to subsequently implement the necessary changes in moving from the CSA N286-05 version of the code to the CSA N286-12 version, during the current licensing period [41]. This timeframe will facilitate the implementation of N286 changes to the management system, and enable the gap analysis results from the large number of new or revised Regulatory Documents or Standards committed in the 2015 operating licence renewal. Bruce Power has also proposed that in the interim, CSA N286-05 be retained in the PROL to enable it to plan the transition to CSA N286-12, and committed to develop the transition plan and communicate the plan to the CNSC by January 30, 2016 [42]. Bruce Power further stated CSA N286-12 does not establish any significant or immediate new safety requirements that would merit a more accelerated implementation. The gap analysis and the resulting transition plan were submitted to the CNSC [43]. Per [43], the major milestones of the transition plan to N286-12 are as follows:

- 22 January 2016: Discuss all the regulatory actions and the transition plan at the (Corporate Functional Area Manager) CFAM meeting

- 31 December 2016: Revision of CFAM Program Document(s) [with LCH notification requirements to the CNSC] to comply with CSA N286-12 requirements completed.
- 31 March 2017: Revision of CFAM Program Document(s) [that do not have LCH notification requirements to the CNSC] to comply with CSA N286-12 requirements completed
- 31 December 2017: Confirmation that that all impacted documents in the program suite comply with the requirements of CSA N286-12
- 15 September 2018: Verification via a Focused Area Self-Assessment (FASA) that previously identified transition Gaps to meeting the requirements of CSA N286-12 have been addressed and effectively implemented
- 14 December 2018: issue notification to the CNSC regarding state of CSA N286-12 readiness, and, implementation date

This Safety Factor therefore has not performed a code-to-code assessment between CSA N286-05 and CSA N286-12 and will not be performing a clause-by-clause assessment of CSA N286-05, since it is in the current licence and there is a transition plan in effect.

CSA-N290.13-05: CSA-N290.13-05 [31], Environmental Qualification of Equipment for CANDU Nuclear Power Plants, is invoked under Licence Condition 5.3 on *Environmental Qualification Program* of the PROL [1], and therefore Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance. There is not a newer version of this standard and its content was reaffirmed in 2010 and again in 2015. Therefore, as reflected in Table C-1 of the PSR Basis Document [5], no further assessment of CSA-N290.13-05 [31] is necessary in the review of this Safety Factor.

3.3. Regulatory Documents

In addition to the Regulatory Documents identified in the Bruce Power PROL [1] and the LCH [2], other Regulatory Documents identified in Table C-1 of the PSR Basis Document [5] considered for application to review tasks of this Safety Factor are included in Table 2.

Table 2: Regulatory Documents

Document Number	Document Title	Reference	Type of Review
CNSC REGDOC-2.5.2	Design of Reactor Facilities: Nuclear Power Plants	[44]	PCBC
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			

CNSC REGDOC-2.5.2: REGDOC-2.5.2 [44], Design of Reactor Facilities: Nuclear Power Plants, has several clauses that address design practices to facilitate aging management. Table C-1 of the PSR Basis Document [5] indicates the need for an assessment against specific clauses in support of the identified review tasks. The results of this partial clause-by-clause assessment are documented in Appendix B.

3.4. CSA Standards

In addition to those identified in the Bruce Power PROL [1] and LCH [2] the CSA standards Identified in Table C-1 of the PSR Basis Document [5] considered for application to review tasks of this Safety Factor are included in Table 3.

Table 3: CSA Standards

Document Number	Document Title	Reference	Type of Review
CSA-N285.8-15	Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	[45]	HL
CSA-N291-15	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	[46]	PCBC
CSA-N287.1-14	General Requirements for Concrete Containment Structures for Nuclear Power Plants	[47]	CTC/CBC
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			

CSA-N285.8: CSA-N285.8-15 [45] Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors is the third edition of this standard. It supersedes the previous editions, published in 2010 and 2005. The requirements of N285.8 address the specific fitness-for-service evaluation requirements of N285.4, Clause 12. The 2010 version of this standard provided updated methodologies for the assessment of pressure tubes and the 2015 provided additional explanatory information on these methods. Therefore a high-level assessment of the differences among the 2015, 2010 and 2005 versions was conducted and included as Appendix A.3 to this report.

CSA-N291: CSA-N291-15 [46], Requirements for Safety Related Structures for Nuclear Power Plants, provides material, design, construction, fabrication, inspection and examination requirements for CANDU safety-related structures. This is the second edition of CSA-N291. It supersedes the previous edition published in 2008 under the title: Requirements for safety related structures for CANDU nuclear power plants. The title has been changed to reflect a change in scope, from addressing only CANDU reactors to including all types of nuclear power plants. Aspects of this standard related to ageing are assessed in Appendix B (B.3) of this Safety Factor Report, whereas a more comprehensive but high-level review of CSA-N291-15 is addressed in “Safety Factor 1: Plant Design”.

CSA-N287.1-14: CSA-N287.1-14 [47], General Requirements for Concrete Containment Structures for Nuclear Power Plants, relates to and is assessed in “Safety Factor 1: Plant Design”. Aspects of this standard related to ageing are assessed in Appendix B (B.2) of this Safety Factor Report.

3.5. International Standards

As applicable, international guidance considered for application to review tasks of this Safety Factor are included in Table 4.

Table 4: International Standards

Document Number	Document Title	Reference	Type of Review
IAEA SSG-25 (2013)	Periodic Safety Review for Nuclear Power Plants	[48]	NA
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			

IAEA SSG-25: IAEA SSG-25 [48] addresses the periodic safety review of nuclear power plants. Per the PSR Basis Document [5] this PSR is being conducted in accordance with REGDOC-2.3.3. As stated in REGDOC-2.3.3 [3], this regulatory document is consistent with IAEA SSG-25. The combination of IAEA SSG-25 and REGDOC-2.3.3, define the review tasks that should be considered for the Safety Factor Reports. However, no assessment is performed specifically on IAEA SSG-25.

3.6. Other Applicable Codes and Standards

The codes and standards discussed in the previous sub-sections have been determined to be sufficient for the completion of the review tasks of this Safety Factor. Accordingly, additional codes and standards are not considered in this Safety Factor Report.

4. Overview of Applicable Bruce B Station Programs and Processes

Ageing management of Bruce B SSCs is not subject to a single dedicated program but is governed by a cross-functional collection of governance documents. The LCH [2] identifies the following BP programs as important to Licence Condition 6.1, *Fitness for Service*:

- BP-PROG-11.01: Equipment Reliability [36];
- BP-PROG-11.04: Plant Maintenance [49];
- BP-PROG-11.02: On-Line Work Management;
- BP-PROG-11.03: Outage Work Management; and,
- BP-PROG-12.02: Chemistry Management.

The Equipment Reliability program plays a central role in aging management with the Plant Maintenance program and Conduct of Plant Operation being two key interfacing programs. These programs and other relevant interfacing and implementing guidance are discussed in more detail in the sections that follow. Table 5[2] provides an overview of the key Bruce Power documents for Ageing Management. Note that not all lower tier documents listed in Table 5 that support the program necessarily belong to the Equipment Reliability program hierarchy, but may belong to and also serve other programs.

Table 5: Key Bruce Power Documents for Nuclear Power Plant Ageing Management

Level 0	Level 1	Level 2	Level 3
BP MSM 1: Management System Manual [50]	BP-PROG-11.01: Equipment Reliability [36]	BP-PROC-00778, Scoping and Identification of Critical SSCs [51]	BP-PROC-00666, Component Categorization [52]
			DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology [53]

[2] Table 5 lists the key governance documents used to support the assessments of the review tasks for this Safety Factor Report. A full set of current sub-tier documents is provided within each current PROG document. In the list of references, the revision number for the governance documents is the key, unambiguous identifier; the date shown is an indicator of when the document was last updated, and is taken either from PassPort, the header field, or the “Master Created” date in the footer.

Level 0	Level 1	Level 2	Level 3
		BP-PROC-00779: Continuing Equipment Reliability Improvement [54]	BP-PROC-00532, Critical and Strategic Spares [55]
			BP-PROC-00534, Technical Basis Assessment [56]
			BP-PROC-00539, Design Change Package [57]
			BP-PROC-00789, Maintenance Strategy [58]
		BP-PROC-00780: Preventive Maintenance Implementation [59]	BP-PROC-00284, Predictive Maintenance [60]
			BP-PROC-00456, Preventive Maintenance (PM) WO Deferral Process [61]
			BP-PROC-00457, Development and Approval of Predefined [62]
			BP-PROC-00501, Integrated Preventive Maintenance Program [63]
			BP-PROC-00599, Engineering Guidance for Preventative Maintenance Program Support [64]

Level 0	Level 1	Level 2	Level 3
			BP-PROC-00603, Preventive Maintenance Program "Just in Time" (JIT review Process) [65]
			SEC-MSS-00004, Proactive Maintenance Processes [66]
		BP-PROC-00781: Performance Monitoring [67]	BP-PROC-00284, Predictive Maintenance [60]
			BP-PROC-00361, In-service Testing and Inspection to Satisfy CAN/CSA-N287.7-08 Requirements [68]
			BP-PROC-00387, Plant Inspection [69]
			BP-PROC-00893, Fuel and Fuel Channel Program [70]
			DPT-PE-00005, Performance Requirements for Contamination Exhaust Control Filters [71]
			DPT-PE-00008, System/Component Performance Monitoring Plans [72]

Level 0	Level 1	Level 2	Level 3
			DPT-PE-00009, System and Component Performance Monitoring Walkdowns [73]
			DPT-PE-00010, System Health Reporting [74]
			DPT-PE-00011, Component Health Reporting [75]
		BP-PROC-00782: Equipment Reliability Problem Identification and Resolution [76]	BP-PROC-00496, Troubleshooting Plant Equipment [77]
			DIV-ENG-00004, Engineering Evaluations [78]
		BP-PROC-00783: Long Term Planning and Life Cycle Management [79]	BP-PROC-00400, Life Cycle Management for Critical SSCs [80]
			BP-PROC-00533, Obsolescence Management [81]

The Bruce Power methodology for Aging Management is shown in Figure 1 as extracted from Appendix B of BP-PROC-00783, Long Term Planning and Life Cycle Management [79]. This flowchart lists the various processes Bruce Power uses to prevent, detect, and mitigate aging degradation to improve equipment reliability. Note that BP-PROC-00783 [79] does not govern all of the processes listed in Figure 1, and this flowchart is intended to be a roadmap for understanding how these processes relate to this goal.

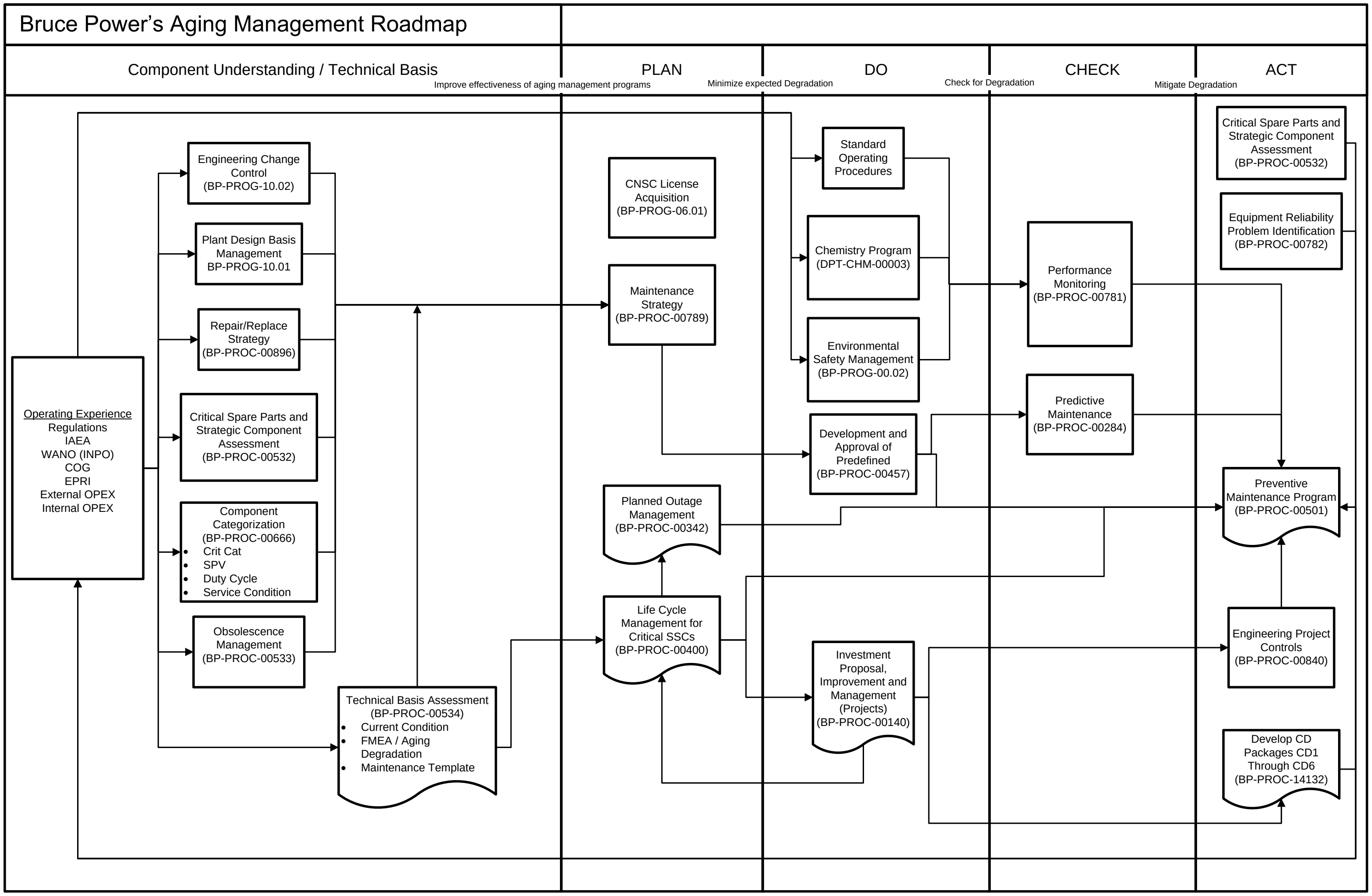

Figure 1: Bruce Power Ageing Management Roadmap

4.1. The Bruce B Equipment Reliability Program

The Equipment Reliability program as defined in BP-PROG-11.01 [36] takes its authority from BP-MSM 1: Management System Manual [50]. The program and its implementing set of procedures are based upon the Institute of Nuclear Power Operations (INPO) Equipment Reliability Process Description (AP-913) [82]. The six implementing procedures of BP-PROG-11.01 [36] are aligned with the 6 sub-processes defined in AP-913 [51]. The six implementing procedures of BP-PROG-11.01 [36] are:

- BP-PROC-00778, Scoping and Identification of Critical SSCs [51]
- BP-PROC-00779, Continuing Equipment Reliability Improvement [54]
- BP-PROC-00780, Preventive Maintenance (PM) Implementation [59]
- BP-PROC-00781, Performance Monitoring [67]
- BP-PROC-00782, Equipment Reliability (ER) Problem Identification and Resolution [76]
- BP-PROC-00783, Long Term Planning and Life Cycle Management [79]

Each of these procedures is described in the following sub-sections, along with the key sub-procedures that specifically relate to ageing management.

4.1.1. Scoping and Identification of Critical SSCs

The scoping and identification of critical SSCs as described in BP-PROC-00778 [51] is an integrated activity that is an input to continuing equipment reliability improvement (covered in procedure BP-PROC-00779 [54]) as well as to the establishment of equipment performance criteria (covered in procedure BP-PROC-00781 [67]).

Components identified as critical (or non-critical if deemed cost effective) undergo preventive maintenance (PM) commensurate with their criticality designation, service conditions, and duty cycle, as outlined in continuing equipment reliability improvement (BP-PROC-00779 [54]). These components and their preventative maintenance are subjected to monitoring requirements established in the procedure on Performance Monitoring (BP-PROC-00781 [67]). All other components are designated as Run-to-Maintenance. BP-PROC-00778 [51] is implemented by the following procedures:

- BP-PROC-00666, Component Categorization [52]; and,
- DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology [53].

BP-PROC-00778 [51] describes the process for identifying SSCs important to maintaining safe, reliable power operation. All aspects of nuclear safety (Reactor Safety, Industrial Safety, Environmental Safety and Radiation Safety) are addressed. The procedure identifies:

- Scoping criteria.
- Functions of SSCs related to safety and reliability.

- Components included in Operation Safety Requirements (OSR) in support of Safe Operating Envelope (SOE).
- Critical structures and components that support these functions.
- Non critical components.
- Run-to-Maintenance components.

Systems important to maintaining safe, reliable power operation include those identified in the safety related system list (see BP-PROC-00169 [83]). Systems important to maintaining safe, reliable power operation will include those identified as systems important to safety as identified through application of DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology [53]. Components important to maintaining safe, reliable power operation include components on the Master Equipment List (MEL) identified as critical or significant to plant operation. This includes:

- Components important to safety in systems important to safety.
- Components that are Single Points of Vulnerability (SPVs).

Components and structures not on the MEL (such as piping, cables and supports) are also reviewed to identify any that are important to maintaining safe, reliable power operation.

SSCs are prioritized in order to optimize safety, reliability, availability, cost and performance within the regulatory framework.

4.1.2. Continuing Equipment Reliability Improvement

BP-PROC-00779, Continuing Equipment Reliability Improvement (CERI) [54], describes the process for development and optimization of the PM technical basis and PM tasks to support a documented PM program for the SSCs identified through application of BP-PROC-0778 [51]. It also identifies major SSC issues for input to BP-PROC-00783, Long Range Planning and Life Cycle Management Plans [79]. BP-PROC-00779 [54] is implemented by the following subordinate procedures:

- BP-PROC-00532, Critical and Strategic Spares [55];
- BP-PROC-00534, Technical Basis Assessment [56];
- BP-PROC-00539, Design Change Package [57]; and,
- BP-PROC-00789, Maintenance Strategy [58].

The technical basis is identified through an industry template or a Bruce Power Technical Basis Assessment (TBA) which contains PM templates that will contain their technical basis.

The CERI process develops a documented ageing management program to avoid SSC degradation or failure, and ensures that continuing adjustments are made to preventive maintenance tasks and frequencies based on operating experience.

The CERI process is aimed at continuous improvement through the identification of alternative strategies, improving existing PM tasks, and adjusting PM frequencies based on reviews of station equipment operating experience. These reviews provide opportunities to delete low value tasks and extend frequencies in order to focus resources on new tasks that are needed and to perform some tasks more frequently, as experience dictates. Input to this procedure includes:

- Critical SSCs identified in BP-PROC-00778 [51];
- Feedback from BP-PROC-00781, Performance Monitoring [67]; and,
- BP-PROC-00783, Long Term Planning and Life Cycle Management [79].

4.1.3. Preventive Maintenance Implementation

BP-PROC-00780 [59] describes the process for carrying out preventive maintenance in support of a continuously improving equipment reliability process in support of BP-PROG-11.01, Equipment Reliability [36]. It is implemented by the following procedures:

- BP-PROC-00284, Predictive Maintenance [60];
- SEC-MSS-00004, Proactive Maintenance Processes [66];
- BP-PROC-00456, Preventive Maintenance (PM) WO Deferral Process [61];
- BP-PROC-00457, Development and Approval of Predefined [62];
- BP-PROC-00501, Integrated Preventive Maintenance Program [63];
- BP-PROC-00599, Engineering Guidance for Preventive Maintenance [64]; and,
- BP-PROC-00603, Preventive Maintenance Program Just in Time (JIT) Review Process [65].

The procedure outlines the interface with the work management system to schedule periodic, predictive and planned maintenance for SSCs on a prioritized/risk informed basis. It also describes the development and use of model work orders to carry out preventive maintenance, and the development and use of a standard set of post maintenance tests to verify important SSC functions and the effectiveness of the maintenance performed.

Preventive maintenance covered by this procedure includes periodic, predictive and planned maintenance. It covers preventive maintenance performed during operation and during outages. Preventive maintenance includes tasks scheduled for components on the Master Equipment List (MEL) (such as pumps, motors, tanks, etc.) and inspection programs carried out for components not on the MEL (such as piping, building structures, feeders, etc.). Consideration is also given to equipment listed within the Operational Safety Requirements (OSR) as part of adhering to the Licence Condition 3.1(i) which requires implementing and maintaining a safe operating envelope in accordance with CSA-N290.15-10, Safe Operating Envelope (SOE) [84] (see DPT-RS-00015, Safe Operating Envelope Gap Assessment [85]).

4.1.4. Performance Monitoring

BP-PROC-00781, Performance Monitoring [67], provides the basis and expectations for the Equipment Performance Monitoring Process at Bruce Power. The SSCs that are included in the performance and condition monitoring program are identified by assessing the criticality of the SSC as well as the OSR. This is done by applying the appropriate screening criteria to the function of the SSC and assessing the impact of SSC failure on plant safety, reliability or economics via BP-PROC-00778 [51].

BP-PROC-00781 [67] describes the process for establishing performance criteria and monitoring parameters for important structures, important system functions and critical components and program performance. It describes:

- Monitoring and trending of system performance;
- Monitoring and trending of component performance;
- Monitoring and trending of program performance;
- Trending of predictive maintenance results;
- Use of operator rounds monitoring;
- Monitoring of Safety System Test (SST) results; and
- Monitoring through Responsible System Engineer (RSE) / Responsible Component Engineer (RCE) walkdowns.

Performance monitoring results are recorded in System Health Reports (SHRs), Component Health Reports (CHRs) or Program Health Reports. The procedure is implemented by the following documents:

- BP-PROC-00284, Predictive Maintenance [60];
- BP-PROC-00361, In service Testing and Inspection to Satisfy CAN/CSA-N287.7 08 Requirements [68];
- BP-PROC-00387, Plant Inspection [69];
- BP-PROC-00893, Fuel and Fuel Channel Program [70];
- DPT-PE-00005, Performance Requirements for Contamination Exhaust Control Filters [71];
- DPT-PE-00008, System/Component Performance Monitoring Plans [72];
- DPT-PE-00009, System and Component Performance Monitoring Walkdowns [73];
- DPT-PE-00010, System Health Reporting [74]; and
- DPT-PE-00011, Component Health Reporting [75].

4.1.5. Equipment Reliability Problem Identification and Resolution

BP-PROC-00782, Equipment Reliability Problem Identification and Resolution [76], describes the problem resolution process, including the interface with BP-PROC-00060, Station Condition Record Process [86] and BP-PROC-00019, Action Tracking process [87]. It describes the process to follow when a critical SSC experiences an unplanned failure or when performance is seen, through Performance Monitoring, to have degraded. This element of the Equipment Reliability process corresponds to the Corrective Action component of AP-913 [51]. Required Corrective Maintenance is executed according to the procedures in BP-PROG-11.04, Plant Maintenance Program [49]. The procedure is implemented by:

- BP-PROC-00496, Troubleshooting Plant Equipment [77]; and
- DIV-ENG-00004, Engineering Evaluations [78].

For an unplanned critical SSC failure, the relevance to nuclear safety is assessed and either an equipment apparent cause or root cause investigation of the degradation or failure is initiated in accordance with BP-PROC-00060 [86]. Corrective actions are determined, including providing feedback to the CERI process.

The need for in-depth analysis of equipment failure is determined by the equipment's importance to plant safety and reliability, as well as the likelihood of failure reoccurrence. When such equipment failure occurs, Plant Engineering uses FORM-14071, Equipment Failure Checklist [88] to ensure the appropriate checks and actions are taken, including:

- Necessary facts surrounding the failure are collected - extent of condition, extent of cause checks are completed as necessary;
- Review and verification of component categorization and PM strategy;
- Assess equipment condition;
- Initiate corrective actions; and
- Initiate required equipment failure causal analysis.

BP-PROC-00782 [76] provides feedback to developing and implementing long-term system or component health improvement plans as part of the Performance Monitoring process. Periodic assessments are made of system, component and program health and vulnerabilities in Health Reports. The system or component health improvement plans are a forward-looking assessment of current problems and future vulnerabilities, providing direction on system or component performance improvement.

The process also interfaces with the Plant Health Committee for prioritization of key equipment problems based on safety, operational impact and station availability (BP-PROC-00559, Station Plant Health Committee [89]). This process also describes how equipment reliability improvement results from a low tolerance for equipment problems and a common station focus to completely resolve key equipment problems.

4.1.6. Long Term Planning and Life Cycle Management

BP-PROC-00783, Long Term Planning and Life Cycle Management [79] enables the development of Life Cycle Management Plans (LCMPs) and the identification and management of obsolescence issues. LCMPs are a significant input to asset management, and are also used as feedback to drive the CERI process (see BP-PROC-00779 [54]). BP-PROC-00783 [79] is implemented by:

- BP-PROC-00400, Life Cycle Management for Critical SSCs [80]; and
- BP- PROC-00533, Obsolescence Management [81].

Asset management as driven by Asset Life Projections & Options (ALP&O) reports facilitates business decisions about capital and Operations & Maintenance (O&M) investments, long-term planning and asset replacement, and maintenance plans and priorities. This drives the following processes:

- Strategic and long-range planning
- Generation planning
- Project evaluation and ranking
- Budgeting
- Plant/fleet valuation
- Aging management

The Bruce Power asset management model is shown in Figure 2.

The Long Term Planning and Life Cycle Management process includes a periodic technical assessment of the plant condition as it relates to its ability to reach its planned end of life. It assesses SSC health and vulnerabilities through an evaluation of aging degradation and an estimate of the remaining service life. External and Bruce Power experience is considered in identifying aging issues. The SSC long-term recommended mitigation options are produced in the Life Cycle Management process, and the approved long-term plan is documented in the Life Cycle Management Plan. If there are major ageing or obsolescence concerns, proactive strategies (e.g., refurbishment/replacement) are to be identified in the Life Cycle Management Process (BP-PROC-00783 [79]) and Obsolescence Management Process (BP-PROC-00533 [81]).

The Life Cycle Management for Critical SSCs (BP-PROC-00400 [80]) provides the basis and expectations for the technical inputs to the Asset Management process. The scope of SSCs to be included in the LCMP process is based on the list of systems important to safety (RD/GD-98 [25]), SPVs, periodic inspection program requirements or whose failure would have a potential impact on plant economics.

The LCMP pulls relevant technical information (e.g., age-related degradation mechanisms, replacement and major overhaul tasks/frequencies, current condition, etc.) from the TBA(s), Performance Monitoring Plan(s), Health Report(s), Performance and Condition Assessments and other data sources and uses this information to document the recommended long-term ageing mitigation options for the subject SSCs.

BP-PROC-00533, Obsolescence Management [81], describes the processes taken to ensure that equipment obsolescence vulnerabilities critical to equipment reliability and plant availability are identified, prioritized and resolved.

The Obsolescence Management process strives to identify and resolve obsolescence issues before they are identified through equipment failure or other emergent circumstances. This is called Proactive Obsolescence Management. The Obsolescence Management Process also provides provisions for Obsolescence issues as they occur during normal work activities.

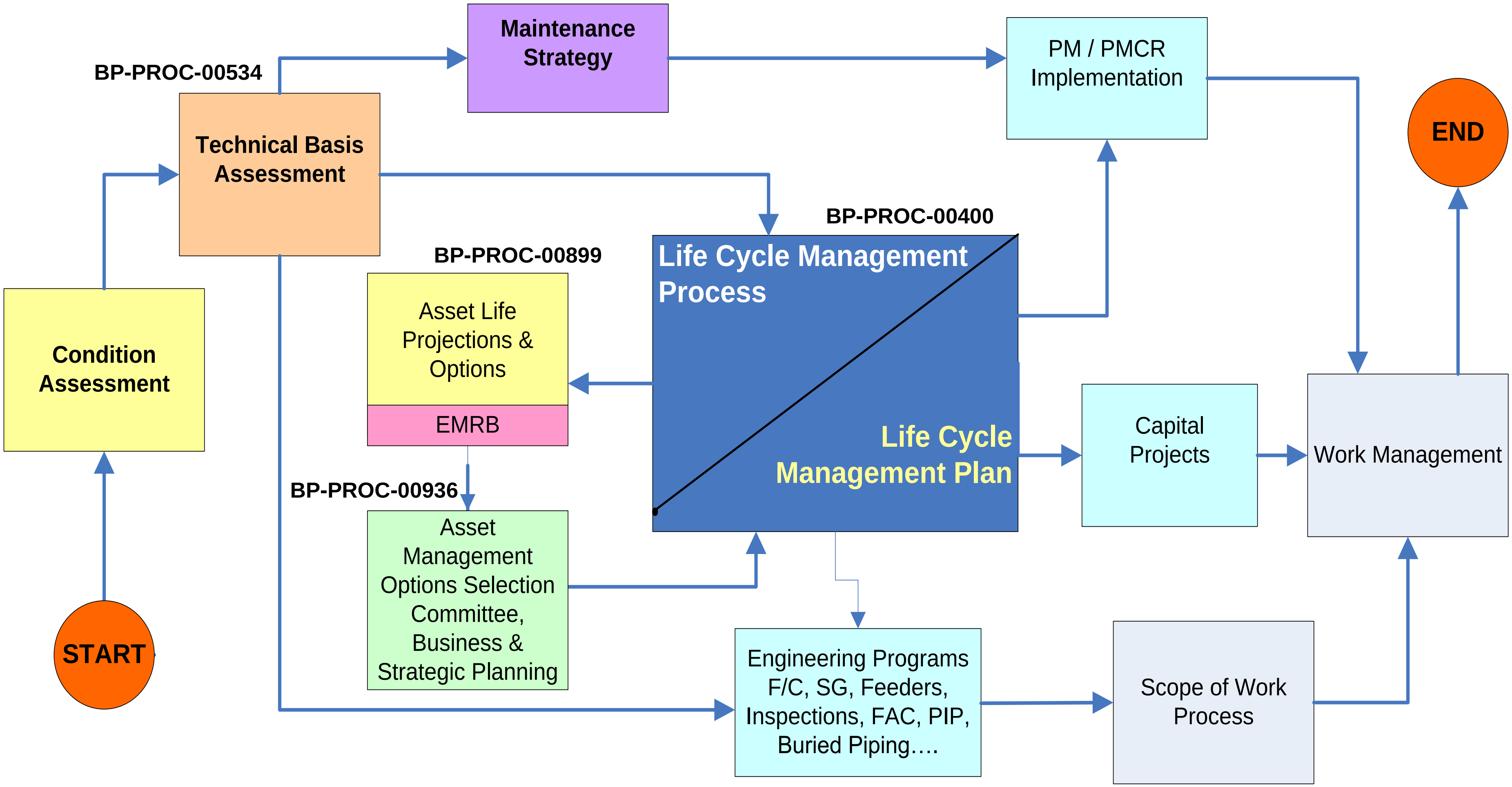

Figure 2: Bruce Power Asset Management Model

4.2. Bruce B Plant Maintenance Program

The Bruce Power Plant Maintenance program supports licence requirements as described in RD/GD-210 [24] and N286-12 [30]. This program is implemented by the following document(s):

- BP-PROC-00695, Maintenance Program and Activities [90];
- BP-PROC-00696, Maintenance Organization [91]; and
- BP-PROC-00699, Maintenance Work [92].

BP-PROC-00695 [90] is of particular importance to aging management since it describes the maintenance program for plant equipment, specifying the following elements:

- What maintenance activities are to be performed on given SSCs and at what frequency/intervals;
- Activities aimed at avoiding, detecting and repairing failures of SSCs;
- Monitoring of the SSCs;
- Maintenance program activity optimization; and
- Record keeping of maintenance performed.

It is written to align and satisfy the expectations set forth by the CNSC in RD/GD-210 [24]. The purpose of the maintenance program is to ensure SSCs function as designed with no unanticipated equipment failures. BP PROG 11.01 [36] and its implementing procedures as described above identify the necessary activities required to monitor and maintain the program.

4.3. Bruce B Conduct of Plant Operations Program

The Conduct of Plant Operations Program, BP-PROG-12.01 [93] provides early warning service of aging related degradation and from this perspective supports the ER and Maintenance programs. A key implementing document that supports the aging management dimension of the Conduct of Plant Operations program is GRP-OPS-00038, Bruce A and B Operations Standards and Expectations [94]. This guidance document not only provides authority to BP-OPP-00001, Bruce B Operating Policies and Principles (OPP) [95], but is in turn supported by the following documents of relevance to aging management:

- BP-PROC-00734, Plant Status Control [96]
- GRP-OPS-00026, Logging Requirements [97];
- GRP-OPS-00047, Operator Routines and Inspections [98], which interfaces with BP-PROC-00781, Performance Monitoring [67];
- GRP-OPS-00001, Operating Memos [99].

The provisions of these documents enable early detection of signs of aging or increased degradation through the daily field inspections by operators, the recording of information that will reveal worsening degradation of equipment condition and performance, and the reporting of an aging related Adverse Conditions through the Event Response and Reporting procedure BP-PROC-00059 [100] and the companion Station Condition Record procedure BP-PROC-00060 [86]. Such reporting may lead to an Equipment Apparent Cause Evaluation (EACE) per BP-PROC-00519, Apparent Cause Evaluation [101] or an Equipment Apparent Root Cause Investigation (ERCI) per BP-PROC-00518, Root Cause Investigation [102].

EACEs and ERCIs are supplemented by Technical Operability Evaluations (TOEs) per BP-PROC-00014 [103] which all feed into the Operational Decision Making procedure as described in GRP-OPS-00030, Operational Decision Making [104] for degradation that impact operability. Furthermore, the licensed Shift Manager performs an Operability Impact check each shift of each relevant SCR to determine operability effect. If deemed necessary an Adverse Condition Monitoring Operating Memo may be prepared per GRP-OPS-00001, Operating Memos [99] to facilitate continued adherence to the OPP. The latter references BP-PROC-00779 [54], Continuing Equipment Reliability Improvement.

4.4. Other Bruce B Programs and Procedures Related to Aging Management

In addition to the procedures related to Equipment Reliability, Plant Maintenance and Conduct of Plant Operations, the following guidance documents are also relevant to this Safety Factor:

- DPT-NSAS-00016, Integrated Aging Management for Safety Assessment, [105];
- BP-PROG-12.02, Chemistry Management, [106];
- DPT-CHM-00003, Control of Chemistry, [107];
- DPT-CHM-00007, Performance Monitoring, [108];
- DPT-CHM-00008, Outage Chemistry Program, [109];
- BP-PROG-10.01, Plant Design Basis Management, [110];
- BP-PROG-10.02, Engineering Change Control, [111]; and
- BP-PROG-10.03, Configuration Management, [112].

In particular, DPT-NSAS-00016 [93] describes how fitness for service inspection/monitoring and safety analysis activities are coordinated to ensure that safety margins are adequate and ageing management issues are addressed. BP-PROG-12.02, Chemistry Management [106], provides governance for control of chemistry (DPT-CHM-00003 [107]), performance monitoring with respect to chemistry control (DPT-CHM-00007 [108]) and the outage chemistry program (DPT-CHM-00008 [109]). The Design Basis Management, Engineering Change Control and Configuration Management programs strongly interface with the ER, Maintenance, and Conduct of Operations programs.

5. Results of the Review Tasks

The results of the review of this Safety Factor are documented below under headings that correspond to the review tasks listed in Section 1.2 of this document. The review tasks assessed in this section have not changed from those listed in Section 1.2.

The objective of the review of ageing 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. The review should evaluate both programmatic and technical aspects.

Sixteen review tasks for Safety Factor 4 are identified in Section 1.2 of this report. Each of the sixteen review tasks was assessed, and the results presented in the following sub-sections.

5.1. Timely Detection and Mitigation of Ageing Mechanisms / Effects

Review Task Interpretation

Review task 1a of Section 1.2 addresses the timely detection and mitigation of ageing mechanisms and/or ageing effects. In addition, Review task 2f of Section 1.2 addresses methods for monitoring ageing and for mitigation of ageing effects, which is closely related to timely detection of ageing effects.

Review Task Assessment

Timely detection and mitigation of ageing are achieved via the PM process [59] and the Performance Monitoring process [67] to continuously confirm effectiveness. These processes are supported by a number of other procedures as described in this sub-section.

The following procedures address the inspection aspect of this review task:

- BP-PROC-00923, Pipe Wall Thinning – Flow Accelerated Corrosion (FAC) [113];
- BP-PROC-00334, Periodic Inspection [114]; and,
- BP-PROC-00825, Buried Piping Inspection Program [115].

BP-PROC-00923, Pipe Wall Thinning – FAC [113], establishes the requirements for the detection of pipe wall thinning due to FAC and the initiation of corrective action at Bruce Power. These activities are performed to maintain piping integrity in order to reduce the risk of injury from piping failures and to ensure that piping systems important to the safe operation of the plant are capable of meeting their design basis requirements.

BP-PROC-00334, Periodic Inspection [114], describes how the requirements for the Periodic Inspection Program of plant SSCs are established and documented through creating, updating and revising the Periodic Inspection Plans and Schedules. The following systems which are subject to periodic inspection under CSA-N285.4 [27] are identified in the Periodic Inspection Plans for Bruce B Units 5 to 8 [116], [117], [118], [119]:

(a) Systems, and systems connected thereto, containing fluid that, under normal conditions, directly transports heat from nuclear fuel and other systems whose failure may result in a significant release of radioactive substances (CSA-N285.4 Clause 3.3.1(a)):

- Primary Heat Transport Main Circuit (includes primary side of vessels)
- Primary Heat Transport Autoclave Circuit
- Primary Heat Transport Feed, Bleed and Relief Circuit
- Primary Heat Transport Storage, Transfer and Recovery Circuit
- Primary Heat Transport Gland Seal Circuit
- Primary Heat Transport Purification System
- Primary Heat Transport Maintenance Cooling System
- Primary Heat Transport D_2O Sampling System
- Primary Heat Transport – Fueling Machine D_2O Auxiliary System

(b) Systems essential for the safe shutdown of the reactor and / or the safe cooling of the nuclear fuel in the event of a process system failure (CSA-N285.4 Clause 3.3.1(b)):

- Emergency Coolant Injection Supply System
- Emergency Coolant Injection System
- Shutdown System 1
- Shutdown System 2
- Moderator System Auxiliary Circuit
- Main Moderator System
- Emergency Boiler Cooling System

(c) Systems, the failure or dislodgement of which could jeopardize the integrity of systems in item (a) or (b) above, or both (CSA-N285.4 Clause 3.3.1(c)):

- Boiler Steam & Feed Water System: Steam Generator and Preheater Secondary Shells and Steam Drums
- For systems subject to periodic inspection under (a) and (b) above: Equipment and Piping Supports and Hangers
- Primary Heat Transport (PHT) Pump Flywheels

Fuel Channel Pressure Tubes, Fuel Channel Feeder Tubes and Steam Generator Tubes are addressed by requirements identified in CSA-N285.4 Clauses 12.0, 13.0 and 14.0. The periodic inspection requirements for these components are specified in the following Bruce Power documents:

Fuel Channel Pressure Tubes (CSA-N285.4 Clause 12.0):

- B-PLAN-31100-00001, Fuel Channel Life Cycle Management Plan (FCLCMP), [120],
- B-PIP-31100-00001, Fuel Channel Periodic Inspection Program, [121]

Feeder Pipes (CSA-N285.4 Clause 13.0):

- B-LCM-33126-00001, PHT Feeder Piping Life Cycle Management Plan, [122]

Steam Generator Tubes (CSA-N285.4 Clause 14.0):

- B-PLAN-33110-00001, Steam Generator and Preheater Life Cycle Management Plan, [123]

Containment boundary components subject to periodic inspection under CSA-N285.5 are identified in NK29-PIP-03642-00001, Bruce B Periodic Inspection Plan for Unit 0 and Units 5 to 8 Containment Components [124].

BP-PROC-00334, Periodic Inspection [114], documents the methods for review, evaluation and disposition of Periodic Inspection findings, as required, and identifies the roles and responsibilities for inspection personnel.

BP-PROC-00825, Buried Piping Inspection Program [115], establishes the process and specifies the requirements to detect and assess degradation in buried piping as a result of its ageing and material degradation due to the effects of related degradation mechanisms, and to initiate corrective action at Bruce Power. These activities are performed to maintain buried piping integrity in order to reduce the risk of the potential impacts to the environment and public confidence in case if unanticipated buried piping failures occur, and to ensure that buried piping systems important to the safe operation of the plant are capable of meeting their design basis requirements until the projected end of life of the generating units/stations.

BP-PROC-00780, Preventive Maintenance Implementation [59], describes the process for carrying out preventive maintenance in support of a continuously improving equipment reliability process. Preventive maintenance includes periodic, predictive and planned maintenance.

The procedure outlines the interface with the work management system to schedule periodic, predictive and planned maintenance for SSCs on a prioritized/risk informed basis. It also describes the development and use of model work orders to carry out preventive maintenance, and the development and use of a standard set of post maintenance tests to verify important SSC functions and the effectiveness of the maintenance performed.

It covers preventive maintenance performed during operation and during outages. Documenting the equipment as found condition is important to a continuously improving equipment reliability process, and BP-PROC-00780 [59] presents the process for capturing information from maintenance personnel on the as-found condition and providing feedback to the RSE/RCE.

Once the PM tasks and frequencies are established per BP-PROC-00779, Continuing Equipment Reliability Improvement [54], in PASSPORT, the Maintenance PM Assessor will generate a PM Identification Requirement (PMIDRQ) from the information provided. BP-PROC-00457, Development and Approval of Predefined [62], provides the process for developing and approving new or changing predefined or model/generated work orders.

A review of the scheduled PM occurs under the Just-in-Time review process, 26 weeks before work week execution as per BP-PROC-00603, Preventive Maintenance “Just-in-Time (JIT)” Review Process [65].

BP-PROC-00501, Integrated PM Program [63], provides the methodology to effectively specify PM activities, achieve ER goals and continuously improve the Bruce Power site PM programs.

BP-PROC-00781, Performance Monitoring [67], provides the basis and expectations for the Equipment Performance Monitoring Process.

Performance Monitoring is supported by BP-PROC-00284, Predictive Maintenance (PdM) [60] which establishes the requirements to implement, maintain and continuously improve the PdM Program integrating various equipment condition monitoring technologies. The program examines and trends critical component data to assess immediate signs of premature ageing via infrared thermography, lubricant analysis, vibration monitoring, and airborne ultrasound. BP-PROC-00284 [60] invokes *inter alia* the following implementing procedures:

- BP-PROC-00323, Predictive Maintenance Lubrication Analysis [125];
- BP-PROC-00762, Predictive Maintenance Ultrasound Inspection Program [126];
- BP-PROC-00768, Predictive Maintenance Infrared Thermography Program [127];
- SEC-RE-00009, Predictive Maintenance Vibration Monitoring [128]; and,
- SEC-RE-00016, Predictive Maintenance Motor Testing Program [129].

Review Task Conclusion

Bruce Power’s Preventive Maintenance process, supported by its various inspection processes and Performance Monitoring process which incorporates significant PdM elements, facilitates timely detection and mitigation of ageing. Bruce Power therefore meets the requirements of this review task.

Bruce Power’s Preventive Maintenance process and supporting processes also include the use of various methods for monitoring ageing and for mitigation of ageing effects, which is related to Review task 2f of Section 1.2. Therefore the assessment documented in this section also confirms the assessment of Review task 2f of Section 1.2, which is documented in Section 5.13.

5.2. Comprehensiveness of Program

Review Task Interpretation

Review task 1b of Section 1.2 requires assessment of the comprehensiveness of the ageing management program, i.e., does it address all SSCs important to safety? This assessment therefore focuses on establishing whether the comprehensiveness of ageing management at Bruce Power is assured by a systematic screening process and criteria.

Review Task Assessment

Part of ensuring a comprehensive ageing management program involves confirming that all SSCs important to safety are identified. The process for identifying SSCs that are important to maintaining safe, reliable operation is described in BP-PROC-00778, Scoping and Identification of Critical SSCs [51]. The first step in the process of identifying the criticality of SSCs is to define the SSC functions that are important to safety and availability. Once the system safety functions have been identified, the critical and non-critical components related to safety can be identified.

The RSE defines the SSC functions that are important to providing safe, reliable power operation by reviewing and evaluating the following:

- SSCs identified in the Safety Related Systems List (BP-PROC-00169 [83]);
- Components identified as SPV (BP-PROC-00666 [52]);
- Systems identified as “important to safety” as defined by the station PRA (DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology [53]);
- Functions identified in the Safety Report, System Design Manuals and station safety analysis;
- Functions and SSCs identified in the OPP and Impairment Manual which indicates special safety systems and safety related systems are safety critical per the Safety Report;
- Regulatory requirements; and
- Environmental Qualification (EQ) Safety Related Component List.

Based on a review of the above information, the important functions of each system are captured in the “Functional Failure Evaluation” section of the System Performance Monitoring Plans (SPMP), per DPT-PE-00008 [72] (see also BP-PROC-00781, Performance Monitoring [67]).

The Functional Failure Evaluation identifies critical system functions, along with equipment or components that provide those functions, and the failure modes that can render the system incapable of meeting performance goals or design functions.

Once the system safety function has been identified, the RSE assesses the criticality categorization (Crit-Cat) for components associated with each important function. BP-PROC-00666, Component Categorization [52], describes the process of categorizing components. A critical component is one whose function is essential to system operation and/or operability (Crit-Cat 1 & 2). BP-PROC-00666, Component Categorization [52] provides detailed definitions of the criticality categories that can be summarized as follows:

- Crit-Cat 1 components are defined as SPVs or components whose failure would result in, for example, an immediate or unavoidable Reactor Trip, Turbine Trip or De rate of > 10% (SPV), or Reactor shutdown;

- Crit-Cat 2 components are those in OSR manual whose failure (and loss of safety function) can be tolerated for a short period of time (consistent with impairment manual);
- Crit-Cat 3 are components not categorized as critical and not classified as run-to-maintenance components. If these SSCs fail they can have a significant impact on meeting economic, radiological, environment and conventional safety objectives. The failure has lower consequences, or are not immediate, leaving time for corrective or compensating actions; and
- Crit-Cat 4 components are run-to-maintenance components, where it is more cost effective to perform corrective maintenance (post failure) than preventive maintenance. Risks and consequences are acceptable without any predictive or preventive maintenance being performed and there is not a simple, cost effective method to extend the useful life of the component.

Components identified as critical (Crit-Cat 1 & 2) or non-critical (Crit-Cat 3) shall (if deemed cost effective for non-critical) undergo preventive maintenance commensurate with their criticality designation, as outlined in BP-PROC-00779, Continuing Equipment Reliability Improvement [54], and will be subject to monitoring requirements established in BP-PROC-00781, Performance Monitoring [67]. Crit-Cat 4 components are deemed “run-to-maintenance” and will not receive any preventive maintenance.

BP-PROC-00666, Component Categorization [52], provides guidance for determining SPV Designation, Service Condition categorization, Duty Cycle categorization and S-98 Equipment Importance designation.

SPV designation is used to identify critical components that, due to a lack of redundancy, represent a greater risk to safe, reliable operation as the plant ages. This designation is one element in assigning the criticality category, determining maintenance tasks, and in prioritizing spare parts needs under procedures BP-PROC-00779, Continuing Equipment Reliability Improvement [54], and BP-PROC-00532, Critical and Strategic Spares [55].

Service Condition and Duty Cycle are required to support maintenance template development and component level PM strategy application (as per BP-PROC-00779, Continuing Equipment Reliability Improvement [54], and BP-PROC-00780, Preventive Maintenance Implementation [59]). S-98 equipment importance is a designation of risk importance is input into many processes including procurement.

Once the categorization is complete, the designations that are documented in PASSPORT are changed using Engineering Change Control Program, BP-PROG-10.02, [130]. This categorization change is controlled by the RSE. The RSE periodically reviews this data and updates are completed as required.

Review Task Conclusion

Through BP-PROC-00778, Scoping and Identification of Critical SSCs [51] and its supporting procedures, Bruce Power has a well-documented systematic process for the selection and classification of SSCs that ensures comprehensive ageing management. Bruce Power therefore meets the requirements of this review task.

5.3. Effectiveness of Operating and Maintenance Policies / Procedures for Managing Ageing of Replaceable Components

Review Task Interpretation

Review task 1c of Section 1.2 addresses the effectiveness of operating and maintenance policies and/or procedures for managing the ageing of replaceable components.

The focus of this review task is on replaceable components, i.e., components for which ageing need not be managed through ongoing mitigation measures and which are by definition not subject to life-cycle management plans. When reaching a prescribed degree of ageing as indicated by some performance measures, these components are replaced by new ones through the normal maintenance processes. Some of these components may be allowed to run to maintenance while others are subject to a performance monitoring and preventive maintenance. If these measures are effective at managing ageing of replaceable components at an acceptable level, trends in both corrective and elective maintenance backlogs should be declining or at least remain steady.

Review Task Assessment

BP-PROG-11.02, On-Line Work Management [131], defines the rules for the management of operations, maintenance and modification work performed during power operation. BP-PROC-00329, On-Line Work Management Process [132], defines the on-line scheduling process which is intended to provide an organized, well-coordinated station collaboration schedule by which fully planned work, system and component tests, corrective maintenance, elective maintenance, preventive maintenance and modifications are systematically identified, scoped, scheduled, executed, monitored and reported. BP-PROC-00329 [132] is the governing document for BP-PROC-00328, Work Prioritization and Approval [133] which allows for work prioritization through the Station Prioritization Matrix provided in its Appendix D.

A review of maintenance backlogs was conducted and submitted to the CNSC under cover of NK29-CORR-00531-11151 [134] in October 2013. The focus of this review was to quantify and assess the maintenance backlogs at Bruce A and B and compare the results with station targets and industry best practices. Further to that review, Bruce Power has had discussions with the CNSC and is responding to actions to improve maintenance planning and scheduling. The CNSC concluded that Bruce Power's documented maintenance program was basically acceptable; however, Bruce Power was not fully meeting the expectations of BP-PROC-00329, On-Line Work Management Process [132], and BP-PROG-11.02, On-Line Work Management [131]. Bruce Power implemented a comprehensive action plan to reduce the backlogs for Elective Maintenance and Preventive Maintenance deferrals. In the Quarterly Field Inspection Report for Q2 2014 (July 1, 2014 to September 30, 2014), CNSC staff concluded that Bruce Power is meeting regulatory requirements related to maintenance, and that CNSC staff have seen recent improvements in the reduction of the maintenance backlogs, although further improvement is still needed [135]. Action Item 1307-4113 [136] was raised requiring regular updates to be provided to the CNSC on progress being made until the backlogs are reduced to a sustainable level that meets industry standards. Action Item 1307-4113 was closed in

November 2015 after the CNSC deemed progress satisfactory (see NK29-CORR-00531-12928 [137]).

The review provided by BP-PROC-00779, Continuing Equipment Reliability Improvement [54], optimizes preventive maintenance by deleting low value tasks or extending frequencies where monitoring fails to show any signs of degradation. This allows resources to be focused on new tasks, or performing tasks more frequently where monitoring indicates equipment degradation or failure. The living maintenance strategy defined in BP-PROC-00779, Continuing Equipment Reliability Improvement [54], provides for adjusting preventive maintenance based on new information, including observed changes in the rate of degradation.

The Temporary Configuration Changes are being managed with a focus on ensuring that they are required for plant configuration. This is effected through BP-PROG-10.03 on Configuration Management [112] and in particular through its implementing procedure BP-PROC-00638, Temporary Configuration Change Management [138].

For equipment with poor maintenance backlog ratings, if the maintenance backlog is affecting the condition of the specific equipment, it will be addressed in the assessment of System and Component Health Reports as indicated in the Bruce Power report B-REP-00701-23SEP2013-057 [139].

Review Task Conclusion

The conclusions of this review task are that Bruce Power is meeting regulatory requirements related to maintenance and there have been recent improvements in reducing the maintenance backlogs. The related action item has been closed and CNSC staff will continue to monitor the maintenance backlog issue as part of ongoing compliance activities. Therefore, Bruce Power meets the requirements of this review task.

5.4. Evaluation and Documentation of Potential Ageing Degradation that May Affect Safety Functions of SSCs Important to Safety

Review Task Interpretation

Review task 1d of Section 1.2 provides for an assessment of the provisions for the evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs important to safety.

In addition, review task 1f of Section 1.2 provides for an assessment of the use of performance indicators to evaluate potential ageing degradation. Review task 1f is assessed in Section 5.6 and contributes, in part, to the assessment of review task 1d.

Review Task Assessment

BP-PROC-00781, Performance Monitoring [67], also describes the process for establishing performance criteria and monitoring parameters for important structures, important system functions and critical components and program performance. It also provides guidance on System/Component/Program Health Reporting. This procedure describes the:

- Monitoring and trending of system performance;
- Monitoring and trending of component performance;
- Monitoring and trending of program performance;
- Trending of predictive maintenance results;
- Use of operator rounds monitoring;
- Monitoring of Safety-Related System Testing (SSTs) results; and
- Monitoring through RSE / RCE walkdowns.

BP-PROC-00781, Performance Monitoring [67], identifies the following data sources that can be used to assist in performance monitoring activities:

- PdM results (as per BP-PROC-00284, Predictive Maintenance [60], i.e., completion notes/codes from PM Work Orders (WOs) in PASSPORT);
- Safety System Testing (SST) results (i.e., review of all completed SSTs which can be obtained from Reactor Safety (RS));
- All WOs against the system/component group (captured through PASSPORT);
- All SCRs against the system/component group (captured through E Suite/ PASSPORT);
- All Small/Capital projects against the system/component group (captured through "Small Projects List" and "Projects Group/PMC");
- System walkdown records to be recorded and filed with the RSE/RCE as per DPT-PE-00009, System and Component Performance Monitoring Walkdowns [73];
- Operator rounds as per GRP-OPS-00047, Operator Routines and Inspections [98];
- Inspection results from PASSPORT and/or Resident Inspection;
- Issues/actions raised by Duty Engineering Manager (DEM);
- As Found Condition reports ("As Found Condition Codes" captured via PM Completion Module in PASSPORT);
- Shift/Outage Logs; and
- Monitoring software (Plant Information, Meridium, Ventyx/IKS Software suite, Smart Signal, etc.).

Data from the various performance and conditioning monitoring data sources listed above are used for the evaluation of SSC performance and the results of such evaluation may identify potential ageing degradation. Once aging degradation has been identified, the affected SSC may be the subject of an Adverse Condition Monitoring Operating Memo prepared in accordance with GRP-OPS-00001 [99]. The operating memo will specify enhanced monitoring requirements, align organizational support, identify contingency needs, and detail expanded communication requirements.

The results of monitoring and trending activities are to be captured within the System/Component/Program Health Reports (SHR/CHR) by the associated RSE/RCE/RPE as per the intervals established in the System Health Reporting procedure, (DPT-PE-00010 [73]) or Component Health Reporting procedure (DPT-PE-00011 [75]). These reports are generated and stored in System IQ, Component IQ and Program IQ, respectively.

If monitoring/trending indicates that the SSC performance has degraded then the RSE/RCE/RPE outlines corrective actions and the strategy to improve system/component/program health through the System/Component Health Improvement Plans (SHIPs/CHIPs) which are presented to the Station Plant Health Committee (SPHC) for approval (refer to BP-PROC-00782 [76], ER Problem Identification and Resolution).

The RSE/RCE/RPE can determine if there is degraded performance by comparing their monitoring/trending results within their SHR/CHR, against the SPMP/Component Performance Monitoring Plan (CPMP). The criteria for degraded performance are:

- Performance criteria have not been met (as per the SPMP/CPMP);
- Trends from as found equipment condition information indicate that the rate of component degradation is worse than expected ("As Found Condition Codes" are prescribed in BP-PROC-00780 Preventive Maintenance Implementation [59]); or
- Conditional/dynamic data monitoring indicates a degrading trend (from RSE/RCE/RPE analysis).

If the results from monitoring/trending activities have identified a degraded SSC condition, the RSE/RCE determines if an SCR is required in accordance with the guidance provided in BP-PROC-00782, Equipment Reliability Problem Identification and Resolution [76].

Ageing of the following major SSCs, which impact on safe and reliable operation of the plant, is discussed in Section 5.9:

- Fuel Channels
- Primary Heat Transport Feeder Piping
- Steam Generators and Pre-Heaters.

In 2009 a CNSC report on the application of the CNSC Risk-Informed Decision Making (RIDM) process to Category 3 CANDU Safety Issues (CSIs) [140] identified 16 Category 3[3] issues of which the following related directly to aging:

- CI1:[4] "Fuel Channel Integrity and Effect on Core Internals";
- PF19: "Impact of Ageing on Safe Plant Operation"; and,

[3] Category 1: The issue has been satisfactorily addressed in Canada.
Category 2: The issue is a concern in Canada - appropriate measures are in place to maintain safety margins.
Category 3: The issue is a concern in Canada - measures are in place to maintain safety margins, but the adequacy of these measures needs to be confirmed.

[4] Ageing of the fuel channels impacts on safe and reliable operation of the plant.

- GL3: “Ageing of Equipment and Structures“.

The CNSC report also identified Risk Control Measures (RCMs) for each CSI. The RCMs for CI1 are:

- Document and implement an integrated Fuel Channel Ageing Management Plan (FCAMP); and,
- Improve pressure tube ageing management program to ensure that the consequences of ageing fuel channel integrity are adequately managed, and that the appropriate information is collected to support the safety analysis assumptions related to pre-accident pressure tubes characteristics.

The RCM for PF19 is:

- Document and implement an Integrated Ageing Management Program (AMP) that ensures plant ageing mechanisms are identified in all safe operating limits, and collects information appropriate to confirm safety analysis assumptions.

The RCMs for GL3 are:

- Document and implement an Integrated AMP;
- Improve ageing management programs to ensure that the consequences of ageing on systems important to safety are adequately managed, and that the appropriate information is collected to support safety analysis assumptions; and
- Complete condition assessment in the context of plant life extension projects.

Bruce Power addressed the RCMs associated with these CSIs and requested reclassification of these issues from Category 3 to Category 2 in December 2012 [141]. In April 2013, CNSC staff reclassified PF19 from Category 3 to Category 2 [142]. In October 2013, CI1 was reclassified to Category 2 [143], and in April 2014, GL3 was reclassified to Category 2 [144].

Review Task Conclusion

The System/Component/Program health evaluation and reporting provisions of BP-PROC-00781, Performance Monitoring, [67] provides a sound foundation for the evaluation and documenting of ageing degradation that may affect the safety functions of SSCs. Bruce Power meets the requirements of this review task.

5.5. Management of the Effects of Ageing on those Parts of the Plant that Will be Required for Safety When the Nuclear Reactor has Ceased Operation

Review Task Interpretation

Review task 1e of Section 1.2 addresses 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.

Review Task Assessment

BP-PROC-00781, Performance Monitoring [67], provides the basis and expectations for the BP Equipment Performance Monitoring Process. The scope of which SSCs are included in the performance and condition monitoring program is identified by assessing the criticality of the SSC. This is done by applying the appropriate screening criteria to the function of the SSC and assessing the impact of SSC failure on plant safety, reliability or economics via BP-PROC-00778, Scoping and Identification of Critical SSCs [51].

Bruce B systems and their relative placement in the hierarchy of importance in the definition of the scope of the performance and condition monitoring program are included as Appendix B to BP-PROC-00781, Performance Monitoring [67]. Components and programs scoped into the performance monitoring program are identified in Appendix C to BP-PROC-00781, Performance Monitoring [67].

The lists of systems and components provided in Appendices B and C of BP-PROC-00781, Performance Monitoring [67], include SSCs that will be needed after operation has ceased such as the Irradiated Fuel Bays and Systems.

Review Task Conclusion

The assessment above indicates that Bruce Power addresses 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.

5.6. Performance Indicators

Review Task Interpretation

Review task 1f of Section 1.2 of SSG-25 [48] provides for an assessment of the use of performance indicators to evaluate potential ageing degradation.

This assessment of this review task also contributes, in part, to the assessment of Review task 1d of Section 1.2 (evaluation and documentation of potential ageing degradation that may affect safety functions of SSCs important to safety), which is documented in Section 5.4.

Review Task Assessment

As discussed in Section 5.4, BP-PROC-00781, Performance Monitoring [67], describes the process for establishing performance criteria and monitoring parameters for important structures, important system functions and critical components and program performance. It also provides guidance on System/Component/Program Health Reporting.

As described in BP-PROC-00781, Performance Monitoring [67], the RSE establishes System Performance Criteria and Monitoring Parameters for their system by capturing the functions important to safety, as identified in accordance with BP-PROC-00778, Scoping and Identification of Critical SSCs [51]. The RSE then identifies the critical and non-critical components for these functions.

Once this is complete, the RSE documents, within the Performance Monitoring Equipment List (PMEL), all the monitoring parameters from the PMs generated from the CERI process (BP-PROC-00779 [54]) for the critical and non-critical components. The PMEL is a list of equipment and system/component performance indicators to be monitored, trended, and analyzed by the System or Component Engineer. This provides the monitoring baseline for the RSE.

Similarly, the RCE or Responsible Program Engineer (RPE) establishes component performance criteria and monitoring parameters for the component group they are responsible for, and documents, within the PMEL, all the monitoring parameters from the PMs generated from the CERI process (BP-PROC-00779 [54]) for the critical and non-critical components identified in PASSPORT, Meridium or the Plant IQ software suite. This provides the monitoring baseline for the RCE/RPE.

The RSE/RCE/RPE performs a "Functional Failure Evaluation", as per DPT-PE-00008, System/Component Performance Monitoring Plan [72], of these components to determine on a functional basis how failure of the component will affect system performance. This analysis captures the degradation mechanisms, as well as the remedial actions.

There is no single performance indicator for ageing. Rather, a number of performance indicators that are related to, but not unique to, ageing are collectively monitored for trends that may be indicative of changes in ageing degradation rates. Table 6 presents a snapshot of some commonly used ageing-related performance indicator results for Tier 1 systems, as extracted in from the System Health Reports for Q4 of 2015. The Performance Indicators identified in Table 6 demonstrate that ageing-related indicators are included in performance monitoring. Depending on the specific SSC, additional performance indicators are monitored, trended and analyzed. For example, the number of SCRs and Operating Memos are monitored for some systems.

If monitoring/trending indicates that the SSC performance has degraded then the RSE/RCE/RPE outlines corrective actions and the strategy to improve system/component/program health through the System/Component Health Improvement Plans (SHIPs/CHIPs) which are presented to the Station Plant Health Committee (SPHC) for approval (refer to BP-PROC-00782, ER Problem Identification and Resolution [76]).

If results from monitoring/trending activities have identified a degraded SSC condition, the RSE/RCE determines if an SCR is required in accordance with the guidance provided in BP-PROC-00782, ER Problem Identification and Resolution [76]. Such an SCR may in turn lead to a TOE or Impairment Manual action to ensure continued adherence to the OPP.

With respect to safety related structures, NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures [145], describes the relevant inspection program to assure structural integrity.

In addition to the above monitoring/trending activities, CNSC REGDOC-3.1.1 sets out reporting requirements for nuclear power plants, including reporting on ageing related metrics such as pressure boundary degradation, plant reliability and preventive maintenance. These metrics, while not explicitly related to ageing management, are indicative of the effectiveness of ageing management.

CNSC REGDOC-3.1.1 [23] is listed as Condition 3.3, Reporting Requirements, in the PROL [1], and therefore Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL. In compliance with CNSC REGDOC-3.1.1, Bruce Power submits quarterly reports on pressure boundary degradation, annual reports on risk and reliability, and quarterly safety performance indicator reports, including preventive maintenance completion ratios.

Table 6: Ageing Related Performance Indicators for Tier 1 Systems, as Extracted from System Health Reports

G=Green, W=White, Y=Yellow, R=Red

Performance Indicator	Functional Failures (U5/U6/U7/U8)		Maintenance backlog (U5/U6/U7/U8)					Operational Challenges (U5/U6/U7/U8)	
	No. of functional failures	Outstanding functional failure corrective actions	Online Deficient Maintenance Backlog	Shutdown Deficient Maintenance Backlog	Online Corrective Maintenance Backlog	Shutdown Corrective Maintenance Backlog	Predefines – total of late and deferred	Open TOE items	CNSC REGDOC-3.1.1 reportable events
SDS1	R/G/R/Y	G/G/G/G	R/R/W/Y	G/G/W/W	G/G/G/G	W/G/G/W	R/R/R/R	G/G/G/G	G/G/G/G
SDS2	G/W/G/Y	G/G/G/G	R/R/R/Y	W/G/G/G	W/G/G/G	G/G/Y/G	R/R/R/R	G/G/G/G	G/G/G/G
Negative Pressure Containment	G/G/G/G	G/G/G/G	Y/R/G/Y	G/G/G/G	G/G/G/G	G/G/G/G	G/G/G/G	G/G/G/G	G/G/G/G
Airlocks, transfer chambers and bulk heads	G/G/R/G	G/G/G/G	Y/W/W/G	G/G/G/G	G/G/G/G	G/G/W/W	W/W/R/G	G/G/G/G	G/G/G/G
Feed, Bleed, Relief and Storage Recovery	G/G/G/G	G/G/G/G	W/R/R/Y	Y/W/Y/Y	G/G/G/G	G/G/G/G	R/R/R/R	Y/Y/Y/Y	G/G/G/G
Emergency Coolant Injection UNIT 0B	G	G	R	G	Y	G	R	G	Y
Emergency Coolant Injection – UNITS 5,6,7,8	G/G/G/G	G/G/G/G	Y/W/R/W	G/W/R/W	G/G/G/G	W/G/Y/W	R/R/R/R	Y/Y/Y/Y	G/G/G/G
Service Water	G/G/G/G	G/G/G/G	R/R/R/R	R/R/R/R	R/R/R/R	G/G/Y/R	R/R/R/R	G/G/G/G	G/G/G/G
Instrument and Service Air	G/G/G/G	G/G/G/G	R/R/R/R	R/G/Y/R	G/Y/G/G/	Y/R/R/R	R/R/R/R	G/G/G/G	G/G/G/G
Powerhouse Heating and Ventilation	Y/G/G/W	G/G/G/G	R/R/R/R	G/G/G/G	R/R/R/R	G/G/G/G	R/R/R/R	G/G/G/G	G/G/G/G
Powerhouse Air Conditioning	G/R/G/G/	G/G/G/G	R/Y/R/Y	-	R/R/Y/R	-	R/R/Y/W	G/G/G/G	G/G/G/G
Control Room	W	G	R	-	R	-	R	G	G

Performance Indicator	Functional Failures (U5/U6/U7/U8)		Maintenance backlog (U5/U6/U7/U8)					Operational Challenges (U5/U6/U7/U8)	
	No. of functional failures	Outstanding functional failure corrective actions	Online Deficient Maintenance Backlog	Shutdown Deficient Maintenance Backlog	Online Corrective Maintenance Backlog	Shutdown Corrective Maintenance Backlog	Predefines – total of late and deferred	Open TOE items	CNSC REGDOC-3.1.1 reportable events
and Chilled Water Systems UNIT 0B									
Main Heat Transport Circuit, Gland Seal Circuit, Feeders, Autoclaves	G/G/G/W	W/G/G/W	R/R/Y/R	R/R/R/R	G/W/G/W	W/Y/Y/R	R/R/R/R	G/G/G/G	G/G/G/Y

Review Task Conclusion

The process described in BP-PROC-00781, Performance Monitoring [67], for establishing performance criteria and monitoring parameters for important structures, important system functions and critical components incorporates the use of performance indicators to evaluate potential ageing degradation. Bruce Power meets the requirements of this review task.

5.7. Record Keeping

Review Task Interpretation

Review task 1g of Section 1.2 addresses record keeping in support of ageing management.

BP-PROC-00780, Preventive Maintenance Implementation [59], BP-PROC-00781, Performance Monitoring [67], and BP-PROC-00782, ER Problem Identification and Resolution [76], are interfacing procedures that lead the continuous improvement process. The recordkeeping that forms part of these processes and the use of that recorded data are essential for ageing management. The assessment of this review task therefore focuses on the nature of records associated with preventive maintenance, performance monitoring and equipment reliability problem identification and resolution.

Review Task Assessment

BP-PROG-03.01 is Bruce Power's Document Management Program [146] to ensure that the preparation, distribution and maintenance of documents are controlled. It covers Controlled Documents which have a defined revision control process, as well as Records which contain information needed to meet business or regulatory requirements.

BP-PROC-00068, Controlled Document Life Cycle Management [147], defines the process of managing the life cycle of Bruce Power Controlled Documents and BP-PROC-00098, Records Management [148], describes the process for managing records at Bruce Power. The storage and retrieval of Records is governed by BP-PROC-00972, Records Retrieval and Secure Storage [149], including responsibilities for process definition, administration and ongoing oversight and monitoring of the storage and retrievable of records.

BP-PROG-03.02 is Bruce Power's Information Technology Program [150]. It defines how Bruce Power manages information technology, including the process to plan and organize, acquire and implement, deliver and support, monitor and evaluate, and govern information technology.

BP-PROC-00124, IT Solutions – Enterprise Asset Management Support [151], describes the process for management and administration of PASSPORT and related systems, including Content Server, Meridium and E-Suite.

Recordkeeping is an essential part of ageing management at Bruce Power. As such, the above programs and processes are used in conjunction with the procedures on Preventive Maintenance Implementation (BP-PROC-00780 [59]); Performance Monitoring, (BP-PROC-00781 [67]); and, ER Problem Identification and Resolution, (BP-PROC-00782 [76]), which are interfacing procedures that lead the continuous improvement process.

BP-PROC-00780, Preventive Maintenance Implementation [59], describes the process for carrying out preventive maintenance in support of a continuously improving equipment reliability process. Preventive maintenance includes periodic, predictive and planned maintenance. A Preventive Maintenance template is a documented maintenance strategy for a particular component type that lists significant failure modes, possible indications of degradation and recommended condition-based or time-based Preventive Maintenance, as well as monitoring and failure finding tests or inspections. The Preventive Maintenance template identifies planned, periodic, and predictive tasks and frequencies, on a structure or component basis, for Category 1, 2 or 3 components. Technical Basis Assessments (BP-PROC-00534 [56]) are conducted to support development of Preventive Maintenance templates. TBAs are developed in the Bruce Power controlled “TBA” WORD template in the format of B-TBA-USI-XXXXX, where “X” represents a sequence number, and are treated as Controlled Documents in Bruce Power’s PASSPORT system. Maintenance templates are documented in the associated Technical Basis Assessments.

BP-PROC-00781, Performance Monitoring [67], provides the basis and expectations for the BP Equipment Performance Monitoring Process. The scope of which SSCs are included in the performance and condition monitoring program is identified by assessing the criticality of the SSC. This is done by applying the appropriate screening criteria to the function of the SSC and assessing the impact of SSC failure on plant safety, reliability or economics via BP-PROC-00778, Scoping and Identification of Critical SSCs [51].

Bruce B systems and their relative placement in the hierarchy of importance in the definition of the scope of the performance and condition monitoring program are included as Appendix B to BP-PROC-00781, Performance Monitoring [67]. A table of components and programs scoped into the performance monitoring program has been included as Appendix C to BP-PROC-00781, Performance Monitoring [67]. The basis for inclusion is a combination of regulatory requirements, the application of external and internal operating experience, and generally recognized industry best practice supported by Engineering Management.

The lists attached as Appendix B and C of BP-PROC-00781, Performance Monitoring [67], may change from time to time as a result of:

- Internal and external operating experience.
- Revisions to the list of systems important to safety.
- Permanent modifications to plant systems, structures and components.
- The results of assessments (e.g., nuclear, radiological, environmental and industrial safety case, ageing, design basis, identification of new SPVs and execution generation risk analysis methodologies).

In these cases, changes to the Performance and Condition Monitoring program are documented by revisions to Appendix B and C of the Performance Monitoring procedure. Reviews of the lists are conducted in accordance with the requirements of BP-PROC-00068, Controlled Document Life Cycle Management [147].

Performance monitoring results are recorded in System Health Reports or Component Health Reports (SHRs/CHRs), which are kept in the Plant IQ/ System IQ/ Component IQ database.

The database also contains System/Component Health Improvement Plans, and can be accessed via the Bruce Power Intranet.

BP-PROC-00781, Performance Monitoring [67], identifies data sources (records) that can be used to assist in monitoring activities. Data records that are required to be kept and to be retrievable are available in PASSPORT and related systems such as Content Server, Meridium and E-suite, for example:

- Preventive Maintenance results (as per BP-PROC-00284 [60], i.e., completion notes/codes from PM Work Orders (WOs) in PASSPORT).
- Work Orders against the system/component group (captured through PASSPORT).
- SCRs against the system/component group (captured through E Suite/ PASSPORT).
- Small/Capital projects against the system/component group (captured through “Small Projects List” and “Projects group/PMC” in Content Server).
- Inspection results from PASSPORT.
- As Found Condition reports (“As Found Condition Codes” captured via Preventive Maintenance Completion Module in PASSPORT).
- Monitoring software (e.g., Plant Information Meridium, Ventyx/IKS Software suite, Smart Signal, etc.).

In addition official unit/crew logs are important sources of operational data that can be used to support monitoring activities. All deficient equipment per unit is captured in the official Authorized Nuclear operator (ANO) log, as well as the overall Crew Log. GRP-OPS-00026 [97] discusses logging requirements of the Conduct of Plant Operations program.

BP-PROC-00782, ER Problem Identification and Resolution [76], describes the problem resolution process, including the interface with the Station Condition Record (SCR) Process (BP-PROC-00060 [86]) and the Action Tracking Process (BP-PROC-00019 [87]). It describes the process to follow when a critical SSC experiences an unplanned failure or when performance is seen, through Performance Monitoring, to have degraded. SCRs that identify conditions that have the potential to impact Operability need to be acted upon promptly, and a determination needs to be made regarding the need for a Technical Operability Evaluation (TOE) per BP-PROC-00014 [103]. Required Corrective Maintenance is executed according to the procedures under BP-PROG-11.04, Plant Maintenance Program [49].

SCRs are stored in and can be retrieved from E-Suite, and TOEs are available from ContentServer. Degraded equipment condition is captured within System/Component Health Reports stored in System IQ/Component IQ, and corrective actions are outlined in System/Component Health Improvement Plans, also stored in System IQ/Component IQ database.

In addition to the Bruce Power’s internal record keeping, CNSC REGDOC-3.1.1 sets out reporting requirements for nuclear power plants, including reporting on ageing related metrics such as pressure boundary degradation, plant reliability and preventive maintenance. These metrics, while not explicitly related to ageing management, are indicative of the effectiveness of ageing management. CNSC REGDOC-3.1.1 [23] is listed as Condition 3.3, *Reporting*

Requirements, in the PROL [1], and therefore Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL. Reports submitted to the CNSC in compliance with CNSC REGDOC-3.1.1 [23] are retained in Bruce Power's PASSPORT system.

Review Task Conclusion

The assessment above indicates that ageing management at Bruce Power includes the generation and keeping of numerous records in the form of data recorded throughout the complete scope of operations and maintenance activities. Bruce Power meets the requirements of this review task.

5.8. Ageing Management Methodology

Review Task Interpretation

Review task 2a of Section 1.2 addresses Ageing Management Methodology, as described in NS-G-2.12, Ageing Management for Nuclear Power Plants [152]. NS-G-2.12 [152] provides high level requirements for Aging Management of NPPs under the following headings:

- Proactive Strategy for Aging Management
- Aging Management in Operation
- Management of Obsolescence
- Review of Aging Management for Long Term Operation
- Interfaces with other Technical Areas

The CNSC's regulatory document on Aging Management, CNSC REGDOC-2.6.3 [26], is based in part on NS-G-2.12 [152], and sets out CNSC requirements for managing the ageing of SSCs, arranged under the following headings:

- Proactive Strategy for Aging Management
- Integrated Aging Management.

Therefore this review task is performed by a review against CNSC REGDOC-2.6.3, Fitness for Service: Aging Management [26], with an emphasis on the need for a proactive and integrated methodology.

Review Task Assessment

CNSC REGDOC-2.6.3, Aging Management [26], supersedes RD-334 [37], which was the previous Regulatory Document on Aging Management. As part of Bruce Power's request to reclassify CANDU Safety Issue GL3, "Ageing of Equipment and Structures" from Category III to Category II, Bruce Power performed a gap assessment against RD-334 [37] near the end of 2012 and submitted it to the CNSC in NK29-CORR-00531-10447 [141]. In this assessment a number of gaps were identified, and the CNSC requested in NK29-CORR-00531-10862 [153] that Bruce Power provide more definitive information for the residual Risk Control Measures (RCMs) relating to RD-334 [37] and the closing of identified gaps by providing specific actions and timelines. This information was subsequently provided in NK29-CORR-00531-10745 [154] and is summarized below:

(a) RD-334 [37], Clause 3.1

- Gap: FORM-10700, Design Scoping Checklist [155] does not include service life or ageing.
- Response: As stated in RD-334 [37]: "In design documentation, demonstrate how past relevant generic ageing issues, relevant ageing management experience, and research results are addressed." Therefore, there must be a requirement added to design documentation to research applicable ageing OPEX for design changes. RD-334 [37] asks for mitigating strategies to be added to design documentation to include design features that mitigate the effects of ageing mechanisms. RD-334 [37] also states: "specify required provisions for ageing management in procurement documents for new facilities and SSCs, including documents from suppliers and other contractors". Therefore, there is a need to ensure design and procurement procedures include ageing documentation. Design procedures (including FORM-10700 [155] from BP-PROC-00539 [57]) will be updated to reflect RD-334 [37] Section 3.1.
- Status: This action has been completed per Action Request REGM 28332951-06.

(b) RD-334 [37], Clause 3.1.1

- Gap: No formal feedback loop exists between fitness for service and safety analysis to request inspection and to communicate results of inspections.
- Response: A procedure identifying the required interfaces and feedback between safety analysis and fitness for service is being developed.
- Status: This action has been completed by the issue of procedure DPT-NSAS-00016, Integrated Aging Management for Safety Assessment [93], per Action Request REGM 28332951-05.

(c) RD-334 [37], Clause 3.4.3

- Gap: Possibility of extended shutdown should be included in BP-PROC-00400, Life Cycle Management Plans for Critical SSCs [80].
- Response: RD-334 [37] states: "Extended shutdowns are reactor shutdowns lasting for a period exceeding one year, and exclude shutdowns for regular maintenance outages. During extended shutdowns, SSCs may need to be placed in temporary lay-up or safe-storage states which require supplementary measures and controls to prevent ageing degradation." BP- PROC-00400 [80] will be revised to include instructions to generate a plan for the equipment upon the possibility of an extended shutdown.
- Status: This action has been completed per Action Request REGM 28332951-08.

(d) RD-334 [37], Clause 4.1, Item 3

- Gap: Training: There is minimal guidance for the LCMP owners in support of the effective implementation and oversight of their respective LCMPs.
- Response: A Focus Area Self-Assessment (FASA) SA-ERI-2012-04 [156], on LCMP Effectiveness was completed. This FASA identified the need for training related to LCMPs. Training will be provided to LCMP owners/authors upon the completion of revisions to BP-PROC-00534, Technical Basis Assessment, [56] and BP-PROC-00400 [80].
- Status: This action has been completed per Action Request REGM 28332951-09.

(e) RD-334 [37], Clause 4.4.1, Item 4

- Gap: Anticipated obsolescence issues: BP-PROC-00533, Obsolescence Management [81], provides a process for both ageing and obsolescence which appears to meet this requirement but is not consistent with BP-PROC-00783, Long Term Planning and Life Cycle Management [79].
- Response: RD-334 states: "Understanding aging NPP management processes shall include requirements for the evaluation of the current understanding of ageing for the selected SSCs. The evaluation identifies: anticipated obsolescence issues." Therefore, there is a need to review and tie in BP-PROC-00533 [81] and BP-PROC-00532, Critical Spare Parts and Strategic Component Assessment [55], with the appropriate governing documents (e.g., BP- PROC-00783, Long Term Planning and Life Cycle Management [79], and BP-PROC-00778, Scoping and Identification of Critical SSCs [51]). This will be done via an update to BP-PROG-11.01, Equipment Reliability [36], Appendix A. BP-PROG-11.01 [36] is currently being reviewed and the implementing procedures (including BP-PROC-00783 [79]) will be updated to include reference to the new BP-PROC-00533 [81].
- Status: This action has been completed per Action Request REGM 28332951-10.

(f) RD-334 [37], Clause 4.4.1, Item 7

- Gap: RD-334 [37] states "a list of data needs for assessment of SSC ageing (including any deficiencies in the availability and quality of existing records)". There may be a gap but this could only be identified through an audit.
- Response: An audit to identify any deficiencies in the availability and quality of existing records as per RD-334 [37] Section 4.4.1 Item 7 will be considered.
- Status: This action has been completed per Action Request REGM 28332951-11.

An updated version of the compliance assessment against RD-334 [37] was included in the 2013 interim PSR, and documented in Appendix N of the 2013 Safety Basis Report NK29-CORR-00531-11397 [6]. In April 2014, CNSC staff re-classified CANDU Safety Issue GL3 “Ageing of Equipment and Structures” from Category III to Category II [144].

Bruce Power submitted a plan for transition to CNSC REGDOC-2.6.3, which supersedes RD-334, in December 2014 (NK29-CORR-00531-12158 [38]). Subsequently BP-PROG-11.01, Equipment Reliability [36] and its implementing procedures have been revised.

Review Task Conclusion

Given that all the actions identified above have been closed, and considering Bruce Power's plan for transitioning to full implementation of CNSC REGDOC-2.6.3, Aging Management [26], described in NK29-CORR-00531-12158 [38], it is concluded that ageing management at Bruce Power is in compliance with CNSC REGDOC-2.6.3 [26]. By implication of the arguments put forward in the *Review Task Interpretation* the Bruce Power ageing management methodology is also in compliance with this review task.

5.9. Understanding of Dominant Ageing Mechanisms and Phenomena

Review Task Interpretation

Review task 2b of Section 1.2 provides for an evaluation of the operating organization's understanding of dominant ageing mechanisms and phenomena, including knowledge of actual safety margins[5].

This review task is interpreted as requiring the identification of the dominant ageing mechanisms of current concern for the Nuclear Power Plant (NPP) as a whole and an evaluation of the degree to which Bruce Power understands the nature, progression and factors which influence the degradation rate. This review task is also interpreted as requiring an evaluation of safety margins based on the understanding of ageing mechanisms.

This review task also overlaps with some aspects of the report on Safety Factor 2: Condition Assessment. Specifically, Section 5.2 of the Safety Factor Report on Condition Assessment assesses existing and anticipated ageing processes.

Review Task Assessment

The Bruce Power Equipment Reliability Program, BP-PROG-11.01 [36], and its implementing procedures, i.e., BP-PROC-00778, Scoping and Identification of Critical Components [51]; BP-PROC-00779, Continuing Equipment Reliability Improvement [54]; BP-PROC-00780, Preventive Maintenance Implementation [59]; BP-PROC-00781 Performance Monitoring [67]; and, BP-PROC-00782, ER Problem Identification and Resolution [76], provide an overall understanding of aging mechanisms. Dominant ageing mechanisms are SSC specific. Currently the fitness-for-service of the following is under scrutiny because of their impact on safe and reliable operation of the plant:

- Fuel Channels
- Primary Heat Transport Feeder Piping

[5] BP-PROC-00786, Margin Management [156], describes how Bruce Power manages design and operating margins.

- Steam Generators and Pre-Heaters

These components are assessed in Sections 5.9.1, 5.9.2 and 5.9.3. Understanding of ageing of other components is also discussed, but in lesser detail, in Section 5.9.4.

5.9.1. Understanding of Ageing of Fuel Channels

The Fuel Channel Life Cycle Management Plan (FCLCMP), B-PLAN-31100-00001 [120], and Fuel Channel Condition Assessment (FCCA), B-REP-31100-00003 [157], provide extensive overviews of all the ageing mechanisms that affect the fuel channel components. The fuel channel component most affected by ageing degradation is the pressure tube which is subject to a variety of degradation mechanisms, including changes in material properties and dimensional deformation due to neutron irradiation, as well as susceptibility to crack initiation due to in-service induced flaws. The FCLCMP and FCCA also illustrate the interactions between the various degradation mechanisms, and the FCLCMP addresses the requirements of Clause 12 of CSA-N285.4 [27] through the Fuel Channel Periodic Inspection Program [121]. The degradation mechanisms associated with Inconel X-750 annulus spacer have been studied and addressed by the Fuel Channel Life Management Program (FCLMP).

The Fitness for Service assessments implement both deterministic and probabilistic approaches as permitted in CSA-N285.8-05 [158] and subject to regulatory approval. CSA-N285.8-10 [159] provides guidance on deterministic and probabilistic evaluation of pressure tube degradation mechanisms. CSA-N285.8-15 [45] provides new methodologies in addition to those in the 2010 version. A high-level assessment of Bruce Power's compliance with the 2015 version of the standard is included in Appendix A.3. The assessment concludes that the 2015 version does not have new requirements that affect Bruce Power's degree of compliance with the standard. The assessment shows that in a continued effort to ensure full compliance with the standard Bruce Power has taken the following measures:

- Provided the CNSC with details of a long term approach to fitness for service assessment for pressure tubes, which was accepted by the CNSC who opened Action Item 1407-4775 requesting Bruce Power to report semi-annually on progress;
- Submitted a long term compliance plan to the CNSC for the long term use of CSA-N285.8 for the fitness-for-service assessments (see NK29-CORR-00531-12902 [160]. This plan was accepted by the CNSC (see NK29-CORR-00531-13312 [161]). In support of this compliance plan Bruce Power has been submitting updated deterministic and probabilistic assessments of pressure tube fitness-for-service for Bruce Units 3 to 8;
- Submitted detailed disposition reports on flaws after each inspection using the methods of CSA-N285.8; and
- Submitted annual progress reports on relevant R&D to the CNSC.

Hence, through active participation in the COG Fuel Channel R&D Program and FCLMP and participation in the update of the Canadian Standards Association (CSA) standard, Bruce Power is in the forefront of knowledge related to fuel channel ageing mechanisms and is compliant to CSA-N285.8-10 [159], as outlined in the report (B-REP-31100-00010 [162]) on fuel channel fitness-for-service assessment against CSA-N285.8.

Overall, the degradation mechanisms affecting fuel channels have been identified and assessed, and mitigating actions have been developed, as documented in the Fuel Channel Life Cycle Management Plan, B-PLAN-31100-00001 [120].

5.9.2. Understanding of Ageing of Primary Heat Transport Feeder Piping

The process for PHT Feeder Piping Life Cycle Management is described in BP-PROC-00731, PHT Feeder Piping Life Cycle Program [163]. A detailed account of ageing degradation mechanisms for the PHT Feeder Piping is provided in the Life Cycle Management Plan, B-LCM-33126-00001 [122] that was prepared in accordance with the procedure for the Life Cycle Management of Critical SSCs, BP-PROC-00400 [80], and complies with all requirements of CSA-N285.4-05 [164].

The PHT Feeder Piping LCMP, B-LCM-33126-00001 [122] identifies all feeder piping system components that are subject to inclusion in the life cycle management plan. Degradation mechanisms that contribute to the ageing of these components are identified and the consequences evaluated. The current practices to mitigate or manage the effects of the degradation of feeder piping components through inspection, maintenance, trending, modification, repair or replacement, and research and development are also presented. The following are addressed in detail:

- Identification and description of the applicable stressors, ageing mechanisms and degradation sites that could affect the operating life of feeder piping system components.
- The consequences of the ageing degradation mechanism on the feeder piping system components.
- Identification and review of the current ageing management practices, including details of the Periodic Inspection Program (PIP) and In-service Inspections, monitoring and trending done to date.
- Well established methodology to demonstrate fitness for service and to manage the ageing of feeder piping per DPT-ENG-00019, Disposition of PIP/In-Service Inspection Results [165].
- A description of research and development (R&D) programs, with recommendations for their execution.
- Instrument lines and structural components located in and around the feeder cabinets, due to the relative proximity of these inspections to feeder inspections.

As documented in the PHT Feeder Piping Life Cycle Management Plan, B-LCM-33126-00001 [122], the current feeder piping degradation mechanisms are well understood and managed. Degradation mechanisms include FAC of the internal surface of the feeders which can result in pipe wall thinning, feeder cracking at feeder bends or repaired welds, general corrosion, fretting due to elongation of fuel channels or differential thermal movement and vibration of feeders, deterioration of feeder components, and fatigue due to pressure, thermal cycling or a seismic event. FAC induced wall thinning is the most active and limiting ageing degradation mechanism which affects the fitness for service of the feeders.

5.9.3. Understanding of Ageing of Steam Generators and Pre-Heaters

The Steam Generator and Preheater Life Cycle Management Plan is documented in B-PLAN-33110-00001 [123]. The LCMP defines age related degradation modes, assesses cumulative damage to date, and predicts future risk, due to this damage, on station/unit objectives. The LCMP defines and integrates the actions or counter measures to be performed on, and in support of, steam generators and preheaters, to achieve an operating life consistent with the Bruce Power business plan.

B-PLAN-33110-00001 [123] establishes very specific steam generator and preheater performance objectives and identifies the actions and initiatives that are required to meet these objectives, justifies these actions, measures their success and adjusts the overall plan accordingly. These actions include in-service inspections, testing and surveillance activities, operation and maintenance activities, repairs, modifications, fitness for service assessments, research, development and analytical methods initiatives, and performance monitoring.

The LCMP contained in B-PLAN-33110-00001 [123] satisfies the requirements of BP-PROC-00267, Management of Steam Generator and Preheater Tube Integrity [166].

Overall, the degradation mechanisms affecting steam generators and pre-heaters have been identified and assessed, and mitigating actions have been developed, and documented in the LCMP (B-PLAN-33110-00001 [123]). The greatest challenge is related to circumferential stress corrosion cracking of the Steam Generator (SG) tubes, particularly at the top of the tubesheet. Other degradation mechanisms are not considered to be life limiting.

5.9.4. Understanding of Ageing of Other Components

LCMPs, like the ones discussed above, pull relevant technical information (e.g., age-related degradation mechanisms, replacement and major overhaul tasks/frequencies, current condition, etc.) from the TBAs, Performance Monitoring Plans, Health Reports, and other data sources and use this information to document the recommended long-term mitigation options for the subject SSC. LCMPs are developed for SSCs that meet all of the following criteria:

- Components of Critical Categorization 1 or 2 as identified through application of the Component Categorization Procedure BP-PROC-00666 [52].
- The total value of the SSC type is equal to or greater than $10M (including installation costs).
- The SSC is susceptible to life-limiting failure mechanisms, which can act over the life of the SSC in the form of aggressive and long-term mechanisms.

Table 7 provides the details of the most recent revisions of LCMPs and LCM Option Sheets for SSCs that meet the above criteria. The list was prepared from a document containing LCMP summaries provided to the CNSC in February 2016 [167].

Table 7: List of Current LCMPs for Bruce B

SSC	Document Number	Revision	Date
SMALL PUMPS AND MOTORS	B-PLAN-04610-00001	R000	19-Dec-12
HEAT EXCHANGER AND CONDENSER	B-LCM-04660-00001	R000	15-Apr-14
PRESSURE VESSELS AND TANKS	B-LCM-04670-00001	R000	25-Sep-15
SECONDARY PIPING	B-LCM-04900-00001	R000	6-Feb-15
NUCLEAR PIPING	B-LCM-04900-00002	R000	11-Feb-16
CRITICAL MANUAL ISOLATION VALVE MITIGATING OPTIONS	B-LCM-04940-00001	R000	19-Feb-14
MOTOR OPERATED VALVE- ELECTRICAL	B-LCM-04940-00002	R000	24-Feb-15
BURIED PIPING	B-LCM-04975.32-00001	R001	22-Sep-15
LARGE NUCLEAR AND CONVENTIONAL PUMP MOTORS	B-PLAN-05600-00001	R000	23-Aug-10
SERVICE WATER PIPING	B-LCM-07211-00001	R000	6-Nov-14
CIVIL STRUCTURES	B-PLAN-20000-00001	R000	5-Jul-10
FUEL CHANNEL	B-PLAN-31100-00001	R005	15-Nov-12
CALANDRIA SHIELD TANK ASSEMBLY	B-LCM-31200-00001	R000	30-Sep-14
STEAM GENERATOR AND PRE HEATER	B-PLAN-33110-00001	R004	11-Feb-11
PHT FEEDER PIPING	B-LCM-33126-00001	R000	2-Nov-14
NEGATIVE PRESSURE CONTAINMENT SYSTEM COMPONENTS	B-PLAN-34200-00001	R000	3-Dec-14
LARGE TRANSFORMER GREATER THAN OR EQUAL TO 10 MVA	B-LCM-50000-00001	R000	3-Dec-14
CONVERTERS, RECTIFIERS AND INVERTERS	B-LCM-50000-00002	R000	6-Nov-15
SWITCHGEAR BUSES AND 13.8 KV BUS DUCTS	B-LCM-50000-00003	R000	27-Nov-15
ISOLATED PHASE BUS	B-LCM-51150-00001	R000	23-Oct-15
CIRCUIT BREAKERS	B-LCM-53000-00001	R000	25-Aug-15
MOTOR CONTROL CENTRES	B-LCM-53300-00001	R000	13-Dec-15
QUALIFIED AND EMERGENCY POWER GENERATOR CONTROLS	B-LCM-54000-00001	R000	14-Dec-15
STANDBY GENERATORS	B-LCM-54600-00001	R000	11-Dec-15
250 VDC CLASS I BATTERY BANKS	B-LCM-55100-00001	R000	4-Dec-15
ELECTRICAL CABLES	B-LCM-57000-00001	R000	11-Nov-15
INSTRUMENTATION & CONTROL - INDICATING ALARM METERS (Sheet 0001)	B-LCM-60400-00001	R000	9-Nov-15
INSTRUMENTATION & CONTROL - ELECTRONIC PROCESS CONTROLLERS (Sheet 0002)	B-LCM-60400-00001	R000	9-Nov-15
INSTRUMENTATION AND CONTROL - PRESSURE TRANSMITTERS (Sheet 0003)	B-LCM-60400-00001	R000	12-Jan-16

SSC	Document Number	Revision	Date
INSTRUMENTATION AND CONTROL - SOLENOID VALVES (Sheet 0007)	B-LCM-60400-00001	R000	13-Dec-15
INSTRUMENTATION AND CONTROL - MAIN CONTROL ROOM PANEL COMPONENTS (Sheet 0009)	B-LCM-60400-00001	R000	12-Nov-15
TRAVELING SCREENS AND TRASH BAR SCREENS	B-PLAN-71120-00001	R000	19-Nov-13

5.9.5. Knowledge of Safety Margins

To demonstrate acceptable safety margins to 2019, Bruce Power completed analyses of the accident scenarios most affected by ageing, i.e., Loss of Flow, Small Break Loss-of-Coolant Accident (LOCA), Slow Loss of Regulation (Neutron Overpower (NOP)) and Large Break LOCA (LBLOCA). Ageing of fuel channels, feeders and steam generators and preheaters was accounted for in the analyses, as described below.

Loss of Flow (LOF) and Small Break LOCA (SBLOCA) analyses used aged TUF full system and single channel models representative of conditions in 2019 (or beyond), and incorporated projected pressure tube diametral creep, pipe roughness, steam generator and preheater tube plugging and steam generator and preheater tube fouling. Where applicable, reactor physics datasets also incorporated the effects of Heat Transport System (HTS) ageing to 2019 through the application of a core wide pressure tube diametral creep value. For NOP analysis, critical channel powers (CCPs) were calculated using the TUF code for a thermalhydraulics model with aged conditions representation of 2019. The analysis credited the implementation of 37M fuel to demonstrate adequate safety margins to 2019. For LBLOCA analysis, the physics model used a core wide average pressure tube diametral creep value corresponding to 2019, but the thermalhydraulics model did not account for 2019 conditions since HTS ageing effects are beneficial to LBLOCA consequences.

The analyses were submitted to the CNSC in December 2013 via NK29-CORR-00531-11325 [168] to demonstrate safe operation of the Bruce B units under 2019 aged conditions.

Review Task Conclusion

The focus of the assessment was on the dominant ageing mechanisms of critical SSCs, and although only 3 major LCMPs were discussed, it is clear that the relevant LCMPs contain vast amounts of in-depth information about these mechanisms and the methods to determine their progression at prescribed intervals through inspections and other performance monitoring efforts. Bruce Power therefore meets the requirements of this review task.

5.10. Availability of Data for Assessing Ageing Degradation

Review Task Interpretation

Review task 2c of Section 1.2 addresses the availability of data for assessing ageing degradation, including baseline data and operating and maintenance histories.

Review Task Assessment

BP-PROC-00779, Continuing Equipment Reliability Improvement [54], describes the process for development and optimization of the preventive maintenance technical basis and requisite tasks to support a documented Preventive Maintenance (PM) program for SSCs identified in BP-PROC-00778, Scoping and Identification of Critical SSCs [51], to be a part of the ER program.

This process provides input for many aspects of ageing management to avoid SSC degradation or failure, and ensure that continuing adjustments are made to preventive maintenance tasks and frequencies based on operating experience.

BP-PROC-00534, Technical Basis Assessment [56], describes the process for developing the Technical Basis Assessments (TBA) for component types. The TBA provides a documented baseline for the maintenance strategy of the component type. The baseline is developed by performing a Failure Modes and Effects Analysis (FMEA) and is documented using a maintenance template. The FMEA lists the degradation mechanisms. Mitigating tasks are identified and appropriate frequencies for these tasks are established.

The TBA considers external and internal Operating Experience (OPEX) to aid in understanding active and potential ageing degradation. The maintenance template serves as the baseline for the development and analysis of specific maintenance tasks, as defined in BP-PROC-00780, Preventive Maintenance Implementation [59] and also captured in BP-PROC-00783, Long Term Planning and Life Cycle Management [79].

Documenting the equipment as found condition is important to a continuously improving equipment reliability process, and BP-PROC-00780, Preventive Maintenance Implementation [59], presents the process for capturing information from maintenance personnel on the as-found condition and providing feedback to the RSE/RCE.

BP-PROC-00781, Performance Monitoring [67], describes the process for establishing performance criteria and monitoring parameters for important structures, important system functions and critical components and program performance. This procedure describes the:

- Monitoring and trending of system performance;
- Monitoring and trending of component performance;
- Monitoring and trending of program performance;
- Trending of predictive maintenance results;
- Use of operator rounds monitoring;

- Monitoring of Safety-Related System Testing (SSTs) results; and
- Monitoring through RSE/RCE walkdowns.

The performance criteria and monitoring parameters are obtained from the SPMPs/CPMPs prepared in accordance with DPT-PE-00008, System/Component Performance Monitoring Plan [72] or from TBAs prepared in accordance with BP-PROC-00534, Technical Basis Assessments [56]. Performance monitoring results are recorded in SHRs and CHRs per the intervals established in the System Health Reporting procedure (DPT-PE-00010 [74]) or Component Health Reporting procedure (DPT-PE-00011 [75]).

Degraded performance can be identified by comparing the monitoring/trending results and data within the SHR/CHR against the SPMP/CPMP.

Inspection reports are also available as inputs for assessing ageing degradation as part of life cycle management process described in BP-PROC-00400, Life Cycle Management of Critical SSCs [80]. Inaugural/baseline inspection data are collected in compliance with the requirements of CSA-N285.4 and N285.5, as described in the procedure on Periodic Inspection, BP-PROC-00334 [114]. Data from periodic inspections are also collected, and findings are reviewed, evaluated and dispositioned.

For safety related structures, NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures [145], describes the relevant inspection program to assure structural integrity. An assessment of this program against the requirements of CSA-N291 relevant to ageing is documented in Appendix B.3. This assessment shows compliance with the applicable requirements of CSA-N291 related to ageing, except for Clause 7.3.4 which requires structural components to be subjected to a visual inspection and other methods of examination following any abnormal/environmental condition.

Review Task Conclusion

Ageing management at Bruce Power includes well developed provisions for the systematic definition of data needs, for baselining such data, and for collecting and assessing field data to assess ageing degradation.

One issue was identified against the requirement in Clause 7.3.4 of CSA-N291 related to visual inspection of structural components following any abnormal/environmental condition. This is identified as Issue SF4-1 in Table 10.

5.11. Acceptance Criteria and Required Safety Margins for SSCs Important to Safety

Review Task Interpretation

Review task 2d of Section 1.2 addresses acceptance criteria and required safety margins for SSCs important to safety.

The implication of this review task is that the impact of ageing should be considered and accounted for in the acceptance criteria and required safety margins for SSCs important to safety.

Review Task Assessment

It has always been recognized that HTS ageing can affect safety margins before the end of plant life. From the beginning of reactor operation, the impact of HTS ageing has been monitored and erosion of margins has been addressed for parameters that are measured continuously at Bruce Power. Erosion of margins is addressed through compliance processes, such as corrections to detector calibration factors to address high reactor inlet temperatures, as well as through physical plant changes, such as SG tube internal diameter cleaning, SG chemical cleaning, and SG pressure setpoint reduction.

Furthermore, safety analysis and assessment plays a key role in ensuring that the impact of ageing is considered and accounted for in the acceptance criteria and required safety margins for SSCs important to safety. The SOE is the set of operational limits and conditions which identify the safe boundaries for plant operation and within which the nuclear station must be operated to ensure conformance with the safety analysis. Operational limits and conditions are taken into account in the analysis assumptions and inputs of Part 3 of the Safety Report. Analysis of accidents impacted by ageing is revised to reflect plant conditions applicable to the licence duration and the results used to confirm the adequacy of the operational limits and conditions, and if necessary, derive a more suitable value for use as an operating limit. DPT-NSAS-00016, Integrated Aging Management for Safety Assessment [93], describes how fitness for service inspection/monitoring and safety analysis activities are coordinated to ensure that safety margins are adequate and ageing management issues are addressed. This procedure is aligned with the requirement that data and information be collected to confirm safety analysis assumptions and derived acceptance criteria continue to be met, as outlined in CNSC REGDOC-2.6.3 [26].

Execution of DPT-NSAS-00016, Integrated Aging Management for Safety Assessment [93], requires the use of the LCMPs for the various PHT components to adjust the input parameters for deterministic safety analysis simulation software to predict the impact of ageing on safety margins. It also provides feedback to the LCMP to inform future performance monitoring efforts, so that simulations can be based on realistic information.

More information on the use of deterministic safety analysis to assess the impact of ageing can be found in Safety Factor Report 5, Deterministic Safety Analysis.

Review Task Conclusion

Ageing management at Bruce Power, in collaboration with the Safety Analysis Program, meets the requirements of this review task.

5.12. Operating Guidelines for Controlling / Moderating Rate of Ageing Degradation

Review Task Interpretation

Review task 2e of Section 1.2 focuses on operating guidelines aimed at controlling and/or moderating the rate of ageing degradation.

Review Task Assessment

The Bruce B Operating Policies and Principles [95] outline operating boundaries within which the Bruce B station may be operated safely.

Given the nature of the degradation mechanisms that lead to ageing the operating factor that provides the greatest ability to control and moderate the effects of ageing is plant chemistry since it specifically influences processes like oxidation and corrosion. The plant Chemistry Management program BP-PROG-12.02 [106] ensures that system chemistry control and surveillance is performed routinely, and chemistry requirements are identified and documented appropriately. It provides governance for Control of Chemistry (DPT-CHM-00003 [107]), performance monitoring with respect to chemistry control (DPT-CHM-00007 [108]) and the outage chemistry program (DPT-CHM-00008 [109]). Furthermore, this program ensures that chemistry specifications and analytic capability are established and are aligned with OPEX information and best industry practices, using the latest available technology and while maintaining a robust quality control program.

Other operating factors such as steam generator secondary side pressure and reactor power also impact the rate of ageing degradation. For example, inside diameter fouling of steam generator and preheater tubes as a result of feeder wall thinning has been identified as one of the main contributors to rising Reactor Inlet Header Temperature (RIHT) phenomenon.

Review Task Conclusion

Bruce Power meets the requirements of this review task.

5.13. Methods for Monitoring Ageing and for Mitigation of Ageing Effects

Review Task Interpretation

Review task 2f of Section 1.2 addresses methods for monitoring ageing and for mitigation of ageing effects, which are closely related to timely detection of ageing effects covered under review task 1a.

Review Task Assessment

The Preventive Maintenance Implementation process and Performance Monitoring process, as described in BP-PROC-00780 [59] and BP-PROC-00781[67] respectively, are used to continuously confirm effectiveness of monitoring and mitigation ageing. These processes are supported by periodic and in-service inspection and testing programs in accordance with BP-PROC-00334, Periodic Inspection [114].

BP-PROC-00780, Preventive Maintenance Implementation [59], describes the process for carrying out preventive maintenance in support of a continuously improving equipment reliability process. Preventive maintenance includes periodic, predictive and planned maintenance.

The procedure outlines the interface with the work management system to schedule periodic, predictive and planned maintenance for SSCs on a prioritized/risk informed basis. It also describes the development and use of model work orders to carry out preventive maintenance, and the development and use of a standard set of post maintenance tests to verify important SSC functions and the effectiveness of the maintenance performed.

BP-PROC-00501, Integrated Preventive Maintenance Program [63], provides the methodology to effectively specify PM activities, achieve ER goals and continuously improve the Bruce Power site PM programs.

BP-PROC-00781, Performance Monitoring [67], provides the basis and expectations for the Equipment Performance Monitoring Process. Performance Monitoring is supported by BP-PROC-00284, Predictive Maintenance (PdM) [60] which establishes the requirements to implement, maintain and continuously improve the PdM Program by integrating various equipment condition monitoring technologies. The program examines and trends critical component data to assess immediate signs of premature ageing via infrared thermography, lubricant analysis, vibration monitoring, and airborne ultrasound.

HTS ageing has a significant impact on reactor operation and safety analysis assumptions. The dominant ageing mechanisms in the HTS are associated with pressure tubes, steam generators and feeders. Mitigation options have been developed and actions implemented to manage ageing of these components, including the following:

- Replacement of feeders;
- Selective defuelling of fuel channels to reduce deformation;
- Implementation of modified 37-element (37M) fuel to mitigate the impact of HTS ageing on margins to critical channel power; and,
- Steam generator primary side divider plate sealing skin installation and repairs.

Review Task Conclusion

Bruce Power's Preventive Maintenance and Performance Monitoring Programs supported by its periodic and in-service inspection and testing programs, include the use of various methods for monitoring ageing and for mitigation of ageing effects.

The Bruce Power AMP therefore meets the requirements of this review task.

The programs assessed as part of this review task also incorporate significant PdM elements to facilitate timely detection and mitigation of ageing, which is related to Review task 1a. The assessment of this review task therefore confirms the assessment of Review task 1a.

5.14. Awareness of Physical Condition of SSCs Important to Safety

Review Task Interpretation

Review task 2g of Section 1.2 addresses awareness of the physical condition of SSCs important to safety and any features that could limit service life.

Given that the actual physical condition of SSCs is addressed in Safety Factor Report 2, Actual Condition of SSCs, this review task is interpreted as a requirement to ensure processes are in place to establish the physical condition of SSCs important to safety.

Review Task Assessment

BP-PROC-00383, Performance and Condition Assessment [169], provides the basis and expectations for the performance and condition assessment process at Bruce Power, which supports the Equipment Reliability Program (BP-PROG-11.01 [36]). The scope of SSCs to be included in the condition assessment process is identified through the LCM process as described in BP-PROC-00400, Life Cycle Management of Critical SSCs [80], based on their criticality as determined by the impact of SSC failure on plant safety, reliability or economics. The data and information on plant SSCs, which is evaluated in the condition assessment process, is collected through the Performance Monitoring process as described in BP-PROC-00781 [67].

Review Task Conclusion

Bruce Power's performance and condition assessment process ensures the condition of SSCs is established. Given the review task interpretation above, Bruce Power meets the requirements of this review task.

5.15. Understanding and Control of Ageing of all Materials and SSCs that Could Impair their Safety Functions

Review Task Interpretation

Review task 2h of Section 1.2 addresses understanding and control of ageing of all materials (including consumables, such as lubricants) and SSCs that could impair safety functions.

This review task includes assessment of the management of ageing of materials in storage. For materials in use in SSCs, ageing is managed by the LCMP for the SSC.

Review Task Assessment

Bruce Power has an explicit procedure for managing the shelf life of materials while in storage. Once deployed, the life cycle of these materials is managed as part of the SSC to which they are applied.

BP-PROG-05.01, Supply Chain [170], governs the management of materials in storage, and the implementing procedure is BP-PROC-00262, Warehouse Operations [171]. Section 4.3.2 of this procedure reads "Items and materials with limited shelf life are identified on the Cat ID as established by RPE in BP-PROC-00999, Selection of Item Shelf Life Requirements [172]. Storage and monitoring requirements are implemented and maintained by the First Line Manager, Warehouse – Stock Keeping in accordance with DPT-MM-00007, Control of Item Shelf Life Management [173]."

Materials that have been deployed from stores and that deteriorate while being used fall under BP-PROG-11.04, Plant Maintenance [49], under the category of preventive maintenance as governed by BP-PROC-00501, Integrated Preventive Maintenance Program [63]). Longer term degradation not addressed by routine preventive maintenance is addressed by ageing management under BP-PROC-00400, Life Cycle Management of Critical SSCs [80].

Within BP-PROC-00501, Integrated Preventive Maintenance Program [63], there is very little direct guidance on how to establish periodic maintenance to address degradation of materials. Similarly, BP-PROC-00400, Life Cycle Management of Critical SSCs [80] provides little direct guidance on how to accommodate deterioration of materials that affect safety, but does provide the process to be followed in establishing a life cycle management program for critical SSCs. As such, ageing of materials deployed for use is managed as part of the SSC to which they are applied. Moreover, BP-PROC-00695, Maintenance Program and Activities [90] invokes BP-PROC-00135, Station Rework Program [174], which will identify deteriorated repair parts or material as a "Parts Deficiency" as part of the required Rework Evaluation.

The Steam Generator and Preheater Life Cycle Management Plan, B-PLAN-33110-00001 [123] describes the Steam Generator Tube Testing Program, including material characterizations to establish chemical composition, heat treatment, grain size, toughness and tensile properties. Degradation and integrity assessments of other components such as divider plates and separators are also performed.

Pressure tube material properties undergo in-service changes due to thermal effects, neutron irradiation and as a result of deuterium ingress. The most common material properties utilized in pressure tube fitness-for-service assessments include:

- Delayed Hydride Cracking (DHC) growth rate;
- Threshold stress intensity factor for DHC initiation;
- Fracture toughness; and
- Tensile properties.

Fracture toughness, DHC growth rate and threshold stress intensity factor for DHC initiation, are monitored by testing pressure tubes removed for material surveillance in accordance with the requirements of Clause 12.4 of CSA-N285.4. The pressure tube tensile properties are not required to be measured as per the material surveillance requirements, however, they are also measured as part of the surveillance program.

The 2009 version of CSA-N285.4 with the 2011 Update is included in the PROL [1]. However, the latest version of this standard is CSA-N285.4-14, which includes requirements on the monitoring of fuel channel annulus spacer material properties. This is currently not addressed in Bruce Power processes, as documented in the high-level review presented in Appendix A (A.1). This is identified as Issue SF4-2 in Table 10.

As documented in the Feeder Piping LCMP, B-LCM-33126-00001 [122], material testing of removed feeders has also been performed, in order to characterize the condition of the removed feeders and gain better insight as to the degradation mechanisms, especially near the Grayloc hub/tight radius bend region where wall thinning is more pronounced.

Review Task Conclusion

The assessment above indicates that ageing management at Bruce Power includes management of the shelf life of materials in storage.

For materials in use in SSCs, ageing is managed by the life cycle management plan for the SSC. CSA-N285.4-14 includes requirements on the monitoring of fuel channel annulus spacer material properties which are currently not addressed in Bruce Power processes. This is identified as Issue SF4-2 in Table 10. Otherwise, Bruce Power meets the requirements of this review task.

5.16. Obsolescence of Technology

Review Task Interpretation

Review task 2i of Section 1.2 focuses on the obsolescence of technology used in the nuclear power plant.

Review Task Assessment

NS-G-2.12 [152] defines technological obsolescence as:

Lack of spare parts and technical support; lack of suppliers and/or industrial capabilities.

Technological obsolescence is covered in Bruce Power's Obsolescence Management procedure, BP-PROC-00533 [81], which has been developed to be in compliance with Clauses 5.1 through 5.7 of NS-G-2.12 [152]. The program is aligned with the recommendations of the TOP401 Technological Obsolescence Program which supplements IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL) [175].

In 2011 Bruce Power internally identified obsolescence management as an area for improvement and, together with industry, developed a change management plan to improve obsolescence management practices. The change management plan involved the following:

- Revising BP-PROC-00533 [81] to address proactive and emergent obsolescence issues incorporating industry best practices;
- Process rollout through awareness communication actions and developing department-specific training to discuss required interaction with the new process;
- Creating an Obsolescence Working Solutions Committee (OWSC) to maintain and drive solutions to completion while being overseen by an Obsolescence Oversight Committee (OOC);
- Developing new reporting methods developed to identify and track obsolescence, including the Monthly Portfolio, SPHC Work Order report, and Action Plan report;
- Establishing and baselining the Obsolescence Process Coordinator (OPC) role; responsible for maintaining awareness of all known obsolescence issues for Bruce A and Bruce B as tracked in the Site Obsolescence Lists (SOL);
- Reviewing related procedures to identify impacts resulting from the revised Obsolescence Management Process and identifying revisions that would be required
- Developing both general awareness and site specific training. (Bruce Power was a leading participant in the development of the computer based training for obsolescence management of the Electric Power Research Institute (EPRI)).

The goal of the Bruce Power Obsolescence Management Process is to identify and resolve obsolescence issues before they are found through equipment failures or other emergent circumstances by ensuring that equipment obsolescence vulnerabilities are identified, prioritized and resolved in short term, long term, and life cycle management. BP-PROC-00533 [81] provides:

- An overview of the site Obsolescence Management process and defines roles and responsibilities;
- Guidance for the proactive identification of obsolete equipment;
- Guidance for the prioritization and management of identified obsolete equipment; and
- Defines the Obsolescence Process Coordinator (OPC) position as a central collection point for all Obsolescence Issues across Bruce Power.

Obsolescence Identification involves identifying vulnerabilities before equipment failure or other emergent issue through the following:

- Obsolescence Plant Impact Report Review;
- Site Obsolescence List;
- Plant Engineering Obsolescence Impact Review;
- Procurement / Supply Chain Obsolescence Identification; and
- Utilizing the Proactive Obsolescence Management System (POMS) and Obsolescence Items Replacement Database (OIRD) as input into proactive identification of obsolescence issues.

The Proactive Obsolescence Management System (POMS) is an industry database created to assist plants in identifying and solving equipment obsolescence issues. POMS is a project of the INPO's Nuclear Utility Obsolescence Group (NUOG) which was formed to take ownership of Obsolescence issues facing the Nuclear Industry and in which Bruce Power is an active participant.

Programmatic Obsolescence Identification methods are in place to proactively identify obsolescence risks which pose a significant risk to the station. These methods include:

- Top 100 Items on the SOL;
- Critical Equipment; and
- SPVs with Zero Stock.

Station Demand Obsolescence Identification methods are in place to proactively identify obsolescence risks through:

- Plant Impact Work Order Reports;
    - SPHC Work Orders;
    - Online Work Orders;
    - Outage Work Orders;
- System/Component Health Reports; and,
- Walk-up Requests.

Prioritization occurs through application of the Obsolescence Value Ranking (OVR) to all Equipment IDs to quantify the risk of equipment obsolescence to prioritize obsolescence issues based on the following parameters:

- Plant Impact Importance
- Plant Demand
- Parts Availability

The SOL is a "Living List" that is determined using the OVR and Station Demands (each station has its own unique list in POMS)

Action Plans (APs) are used to document the solution strategy for obsolescence issues and track solutions to completion. APs are developed by the OPC in POMS using a graded approach. The solution strategy outlined within the APs are reviewed and approved by the OWSC. APs are tracked by the OPC on the SOL through completion. All APs are completed and stored within POMS.

Action Requests (ARs) (type OBSE - obsolescence action plan) are created to drive the Action Plans. The Action Request (type OBSE) outlines the issue that is to be resolved and the supporting action(s), if required, are entered into the Action Tracking module as assignments under the AR, per BP-PROC-00019, Action Tracking [87]. This procedure aligns with INPO AP-913 [82] and other EPRI guidance.

The OWSC ensures key stakeholders affected by obsolescence issues coordinate and agree on solution strategies. If a recommended solution path requires inter departmental support, the OOC ensures that necessary endorsements from responsible engineers are obtained.

Ongoing tracking and monitoring of the Obsolescence Management Process includes:

- Tracking and analyzing various criteria on a monthly basis by the OPC;
- Monitoring of an reporting on a range of obsolescence metrics; and
- Reporting to department management on monthly basis.

Using POMS data, Bruce Power has established an executive summary for the monthly station portfolio to provide a one-page status update on important programmatic measures, including:

- Obsolescence status trending and industry comparison;
- Trending of 'ins' and 'outs' and the impact on overall obsolescence;
- Performance metrics for solutions;
- A key performance indicator section that includes indicators for overall obsolescence, critical obsolescence, and SPV obsolescence.

Review Task Conclusion

Bruce Power's governance conforms to the latest recommended industry practices for the management of obsolescence of technology. Bruce Power meets the requirements of this review task.

6. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce B PSR. The following identifies specific aspects of this Safety Factor that are addressed in, or where more detail is provided in, another Safety Factor Report.

- "Safety Factor 2: Actual Condition of SSCs" in Section 5.2, overlaps with this report, specifically in regards to existing and anticipated ageing processes, and Section 5.14 the assessment of the verification of the actual state of SSCs against the design basis. Section 5.9 of Safety Factor 2 also supports the understanding of ageing and implementation of recommendations from condition assessments to improve the Life Cycle Management Plans.
- "Safety Factor 3: Equipment Qualification" in Section 5.2 addresses the process for maintaining environmental qualification for the remainder of station life and promotes the understanding of dominant ageing mechanisms.
- "Safety Factor 5: Deterministic Safety Analysis" in Section 5.3, addresses aspects of ageing that relate to current safety assessments and future updates, as well as assessing the validity of assumptions made in the deterministic safety analysis given the actual condition of the plant.

- "Safety Factor 6: Probabilistic Safety Analysis" in Section 5.5.2, addresses the probabilistic risk assessment (PRA) required for risk-based significance screening criteria used for the Systems Important to Safety Decision Methodology discussed in DPT-RS-00012.
- "Safety Factor 8: Safety Performance" in Section 5.5, addresses maintenance performance and the plan to reduce maintenance backlog.
- "Safety Factor 10: Organization and Administration" in Section 5.4.9, addresses organizational units within Bruce Power. In Section 5.3.3 of the Safety Factor 10 report the control of records of baseline information and operational and maintenance history is also addressed.

7. Program Assessments and Adequacy of Implementation

Section 7 supplements the assessments of the review tasks in Section 5, by providing information on four broad methods used to identify the effectiveness with which programs are implemented, as follows:

- Self-Assessments;
- Internal and External Audits and Reviews;
- Regulatory Evaluations; and
- Performance Indicators.

For the first three methods, the most pertinent self-assessments, audits and regulatory evaluations are assessed. Bruce Power has a comprehensive process of reviewing compliance with Bruce Power processes, identifying gaps, committing to corrective actions, and following up to confirm completion and effectiveness of these actions. While there have been instances of non-compliance with Bruce Power processes, Bruce Power's commitment to continuous improvement is intended to correct any deficiencies.

For the fourth method, the performance indicators relevant to this Safety Factor are provided. These are intended to demonstrate that there is a metric by which Bruce Power assesses the effectiveness of the programs relevant to this Safety Factor.

Taken as a whole, these methods demonstrate that the processes associated with this Safety Factor are implemented effectively (individual findings notwithstanding). Thus, program effectiveness can be inferred if Bruce Power processes meet the Safety Factor requirements and if there are ongoing processes to ensure compliance with Bruce Power processes. This is the intent of Section 7.

7.1. Self-Assessments

Generally, self-assessments are used by functional areas to assess the adequacy and effective implementation of their programs. The results of each assessment are compared with business needs, the Bruce Power management system, industry standards of excellence and regulatory/statutory or other legal requirements. Where gaps are identified, corrective actions are identified and implemented.

The self-assessments:

- Identify internal strengths and best practices;
- Identify performance and/or programmatic gap(s) as compared to targets, governance standards and "best in class";
- Identify gaps in knowledge/skills of staff;
- Identify the extent of adherence to established processes and whether the desired level quality is being achieved;
- Identify adverse conditions and Opportunities for Improvements (OFI); and
- Identify the specific improvement corrective actions to close the performance/programmatic gap.

This section contains information on self-assessments related to procedures. Self-assessments are conducted by the line organization as part of the program for continual improvement.

Self-assessments that are relevant to SFR4 and that have been conducted since 2010 are listed in Table 8 as evidence that program effectiveness is being monitored.

Table 8: Self Assessments Relevant to SFR4 Conducted Since 2010

Assessment Number	Title
SA-BAOP-2010-02	Conduct FASA on Plant Status Control DPTSOAB
SA-MPA-2010-03	Outage Execution – Maintenance Milestones
SA-NSAS-2010-03	Use of OPEX in Fuel Channels Life Cycle Mgt & Life Extension of Fuel Channels
SA-ELCE-2011-02	Aging and Obsolescence Project Review
SA-ELCE-2011-08	Assessing the Interactions Between Departments for Improved Performance in Equipment Reliability
SA-MPR-2011-07	Valve Maintenance
SA-OCP-2011-03	Plant Status Control
SA-RPR-2011-01	Fixed Instrumentation Calibration & Maintenance Processes
SA-RPR-2011-02	Portable Radiation Instrumentation Calibration & Maintenance Process
SA-WMSI-2011-04	Effectiveness of Actions Taken as a Result of CNSC Audit BNPD-2009-AB-009-A1
SA-COM-2012-05	MEL Quality Review

Assessment Number	Title
SA-ERI-2012-04	Assessing Life Cycle Management Plan Effectiveness
SA-ERI-2012-01	PMOG Effectiveness
SA-ERI-2012-03	Predictive Maintenance Integration
SA-MPR-2012-02	FLMs In The Field
SA-MPR-2012-07	Long Range Cycle Planning
SA-ERI-2012-05	Hydraulic Pump Monitoring
SA-MPR-2012-10	FLM Knowledge of Predictive Maintenance Program
SA-MPR-2012-09	Control, Storage, Inspection of Lifting & Rigging Equipment
SA-ERI-2012-02	Mechanical Joint Program
SA-MPR-2012-06	Post Maintenance Testing
SA-ERI-2013-05	Equipment Reliability Performance Review Meeting
SA-OGO-2013-01	Maintenance & Test Equipment (M&TE) Data
SA-OGO-2013-01	A2141 Pilot Assessment
SA-ERI-2013-01	Component Programs
SA-ERI-2013-04	System Engineering Effectiveness
SA-ERI-2013-02	Engineering Program
SA-OGO-2013-03	P06 - A1431 Readiness review
SA-ERI-2013-03	System and Component Performance Monitoring Program Compliance
SA-ERI-2013-07	Station Engineering Training FASA
SA-CHEM-2013-01	Chemistry Quality Assurance/Quality Control Management Standards
SA-ERI-2013-06	Buried Piping Program
SA-ERI-2013-08	PM Program
SA-MPR-2013-06	Foreign Material Exclusion
SA-ERI-2013-08	Effectiveness of ERCOE Implementation
SA-MPR-2013-03	Post Maintenance Testing
SA-CHEM-2014-01	Roles and Responsibilities of Station Chemists
SA-MPR-2014-02	Foreign Material Exclusion
SA-MPR-2014-03	Post Maintenance Testing
SA-CHEM-2014-02	Administrative Level Review
SA-CHEM-2014-03	Chemical Technician RP practices
SA-ERI-2014-01	Review of Data Needs to Assess SSC Aging

Assessment Number	Title
SA-ERI-2014-07	Quality of System Health Reporting
SA-MPR-2014-08	SECNMMM Equipment Capability
SA-ERI-2015-02	Use of Condition Based Maintenance for Scheduling Decision
SA-ERI-2015-04	Alignment of ER Governance Implementation at Bruce A and Bruce B
SA-MPR-2015-09	Inspection and Test Plans
SA-ERI-2015-11	System Performance Monitoring Plan (SPMP) Effectiveness
SA-ERI-2015-12	Solenoid Valves Component Health Reporting Effectiveness
SA-ERI-2015-13	Evaluating Pipe Support Inspection Scope and Resourcing
SA-ERI-2015-14	Evaluating Service Water Piping Inspection Program Scope Execution
SA-ERI-2015-15	Relief Valve Quality Program Evaluation

A subset of the self-assessments listed in Table 9 which are more directly relevant to ageing management, and which were performed more recently, are summarized below.

SA-ERI-2012-04 Assessing Life Cycle Management Plan Effectiveness [176]

This FASA evaluated the effectiveness of the Life Cycle Management Plans in order to determine gaps and major areas for improvement in the revision and implementation of LCMPs.

As a result of the issues identified during this FASA, the following corrective actions were initiated:

- Almost all of the LCMPs need substantial revision to be made consistent with the newer BP-PROC-00400 R001 (now at R002 [80]). Moving forward, it will be necessary to begin revising these LCMPs in order to bring them into compliance with R001 and make them usable for their intended purpose.
- It will be necessary to improve training and, wherever necessary, improve supporting documentation to ensure the RSE/Responsible Component Engineers (RCEs) are aware of their duties and responsibilities and can carry them out effectively. “How” and “when” an RSE/RCE or Owner should interact with their LCMP needs to be clarified. A deeper and more thorough understanding of where everything 'fits together' with the RSE/RCEs and Owners is the best path forward.
- Whether it is listed as an objective on the Task Order Quotation (TOQ) or not (for some it is, and for others it is not), maintenance/inspection/refurbishment timelines incorporating all projections and options up to end of life should be included in an LCMP, since this is the primary function of the LCMP. TOQ contract objectives for future LCMP creations or revisions needs to be made clearer to reflect the goals of the LCMPs.
- Fuel Handling LCMPs (two for Bruce A and two for Bruce B) are still using General Electric (GE) document numbers from their original creation, and must be switched to Bruce Power document numbers and properly added to PASSPORT.

SCR 28305457 was raised to address these actions. All of the assignments associated with this SCR have been completed.

SA-ERI-2013-03 System and Component Performance Monitoring Program Compliance [177]

The purpose of this FASA was to assess the procedural and programmatic compliance to determine if work practices are meeting the requirements described in system and component monitoring programs and procedures.

Eight issues and two opportunities were identified during this FASA. SCR 28409862 was raised to track implementation of the corrective actions and recommendations resulting from these issues and opportunities, as follows:

Issue #1 corrective action: RSEs/RCEs to review System/Component OPEX, Environmental Qualification Assessments, Environmental Qualification Dossiers and plant modifications that are related to their areas of responsibility and incorporate changes in PMP (if required).

Issue #2 corrective action: RSE/RCE to compare their respective SPMP/CPMP to ensure they are in alignment and make corrections as required.

Issue #3 corrective action: RSE/RCE to add specific notes in health reports to directly state that the RSE/RCE has been consulted before issuing of a health report. This can be added as a requirement in the System Health Reporting Procedure (DPT-PE-00010).

Issue #4 corrective action: Revise DPT-PE-00008 to more clearly state what is required in this section; it will then be included with the next SPMP revision on each system.

Issue #5 corrective action: Manager oversight to ensure these sections are included in CPMPs.

Issue #6 corrective action: Enforce the expectation to complete walkdowns as specified in PMPs. Develop method of tracking progress, create improvement plan.

Issue #7 corrective action: Enforce the expectation to document walk downs performed, standardize the process/recording method.

Issue #8 corrective action: Assign individuals to add walkdown tasks to Engineering Work Management System.

Opportunity #1 recommendation: standardize and communicate record keeping requirements at the Section level. Organize Section shared folders.

Opportunity #2 recommendation: Revise DPT-PE-00008 to include specific instructions for EQ Program Inclusion.

All of the assignments associated with SCR 28409862 have been completed.

SA-ERI-2014-01 Review of Data Needs to Assess SSC Aging [178]

The objective of this FASA was to identify and review the data needs required to complete an assessment of SSC aging in accordance with CNSC RD-334 (4.4.1), now CNSC REGDOC-2.6.3 (4.2) [26].

No adverse conditions were identified during this FASA, and one opportunity for improvement was identified (in Section 7.3), as follows:

"Area for improvement identified in Technical Basis Assessment and Life Cycle Management procedures (BP-PROC-00534 and BP-PROC-00400 respectively). Data requirements for an effective aging management program require clarity in these procedures, per the guidance in CNSC REGDOC-2.6.3."

SCR 28462763 was raised to track this opportunity for improvement and the assignment associated with this SCR has been completed.

SA-ERI-2014-07 Quality of System Health Reporting [179]

This FASA assessed the quality of System Health Reports (engineering deliverable) and how effectively the content is communicated to influence decision making. One adverse condition and one opportunity for improvement were identified.

The adverse condition indicates that System Health Reports and their contents are not being adequately communicated to decision makers to obtain the appropriate focus and endorsements. SCR 28452101 was raised to address this adverse condition and has been completed. As a result of this SCR, DPT-PE-00010 System Health Reporting[74] and BP-PROC-00559 Station Plant Health Committee[89] have been revised.

The opportunity for improvement indicates that there are specific and common sections of the System Health Reports that are being prepared to a lower quality standard than should be expected. SCR 28452107 was raised to address this opportunity for improvement. As a result of this SCR, DPT-PE-00010 System Health Reporting [74] has been revised.

SA-MPR-2014-08 SECNMMM Equipment Capability [180]

The purpose of this FASA was to identify critical areas of equipment and technologies within the Mechanical Maintenance Section (under the Central Maintenance Department), in order to determine if replacements or upgrades are needed. The FASA examined the following:

- Procedures or guidance related to asset management,
- Tools, technology and processes used by the site weld and machine shop crews,
- Measuring and drafting technologies associated with reverse engineering,
- Information management systems in support of these programs.

Two adverse conditions were identified (in Section 7.2), as follows:

"The exercise of evaluating equipment and technologies is industry accepted practice, but formal and effective guidance does not exist internally at this time, as it applies to our area." SCR 28451465 was raised to address this issue, and the associated assignment has been completed.

"The small number of related SCRs being input may indicate the SCR program does not appear to be fully utilized to drive programmatic improvements with respect to SECNMMM". SCR 28451469 was raised to address this issue, and the associated assignment is complete.

In addition, the following opportunities for improvement were identified (in Section 7.3):

"Equipment and technologies not currently in use should be sought out, evaluated, categorized and added to a detailed assessment process from time to time. These may come from benchmarking, OPEX, trade shows or other sources. Past OPEX from the last three years indicates that it's difficult to release staff to benchmark and find appropriate places to visit." This has been addressed by SCR 28451474 for which the associated assignment is complete.

"Establishing a comprehensive asset lifecycle management procedure would be helpful, in terms of doing analysis, as well as business planning activities. Much information exists on the web however determining what best suits our needs is a challenge. More evaluation in this area needs to happen before a decision is made, or a method is chosen. It is unclear who in OMS should be responsible to put this in place". This opportunity for improvement was addressed by SCR 28451479 (there are no open assignments).

SA-ERI-2015-02 Use of Condition Based Maintenance for Scheduling Decision [181]

This review was performed as part of the Equipment Reliability Program 2015 Self Evaluation plan. The purpose was to do an interim check on the progress of the Equipment Health initiative's focus on better use of condition monitoring and Condition Based Maintenance (CBM) information as an effective input to maintenance work scheduling decisions.

The review focused on the two main avenues of interlace between the Performance and Condition Monitoring activities conducted by the Station organizations (Operations, Maintenance, Chemistry and Engineering), and the Work Management process for Scoping and Scheduling field work: New Work Prioritization; and, T-26 JIT Review Process, as an input to Work Scoping activities at T-17.

The assessment resulted in the following recommendations:

> Completion of this FASA has indicated good progress has been made at both stations regarding the interface with the New Work Prioritization process. However, there are opportunities to further strengthen the interface at the T-26/T-17 interface point to assist with informed scheduling of planned work.
>
> Use of CBM or Condition Monitoring inputs as a driver for Work Management scheduling decisions is well established as part of the New Work Prioritization process at both Stations. Continued oversight of the use of the COGNOS 3399 report identifying Predicted Failures, coupled with increasing visibility of the associated Work Orders to address the degradation through the Catches, Saves and Misses (CSM) metric presented to the SPHCs is recommended.
>
> Regarding the T26 JIT Review interface, the recommendation is to begin to shift the focus of the T26 meeting's use of CBM or Condition Monitoring inputs more towards the interface with Work Management decisions. To achieve that focus shift, the recommended approach is to realign the Terms of Reference for the Station Engineering Basis Oversight Board (SEBOB) to include a challenge of Time Based PMs scheduled at T26, to find opportunities for re-scheduling PMs based on inputs from CBM or Condition Monitoring activities. An action already exists to update the SEBOB Terms of Reference to achieve this re-aligned focus.

This action was raised under SCR 28487709 which is scheduled for completion in June 2016.

While no Adverse Conditions were found, two Opportunities for Improvement were identified and raised as assignments under SCR 28531165. The assignments are the following:

- Conduct a planned review and effectiveness check of the ongoing utilization of PF work orders as an input to the New Work Prioritization process. This action is to ensure sustainability of the processes that are established at both Stations; and,
- Conduct a planned review and effectiveness check of the revised SEBOB Terms of Reference. This action will be an initial check that the SEBOB challenges of Time Based PM Scheduling are effectively using CBM and Condition Monitoring results as the basis for recommended scheduling decisions at the T17 meetings at both Stations.

The two assignments are due for completion in July 2016 and November 2016 respectively.

SA-ERI-2015-04 Alignment of ER Governance Implementation at Bruce A and Bruce B [182]

A self-assessment exercise was performed, in conjunction with the International Atomic Energy Agency (IAEA) Operational Safety Review Team (OSART) review of Bruce B, to critically assess use of the relevant Bruce Power procedures that are in place to support development of Ageing Management Programs at the Bruce B and Bruce A Stations as Bruce Power prepares for transition to Long Term Operations when the current site licence expires in 2020.

The purpose was to complete an alignment check of procedure implementation as challenged by the international expert assessments conducted during the OSART review against relevant IAEA Safety Guides established as a framework for excellence in managing critical Systems, Structures and Components as Stations transition to Long Term Operations.

The assessment concluded as follows:

> Completion of this FASA has confirmed that implementation of relevant Bruce Power Governance at both Stations is aligned, and is consistent at both Bruce A and Bruce B. Implementation of relevant procedures is consistent at both Stations, and due to the fact that most Asset Management related activities are executed in support of both Stations at the same time, the progress of implementation is largely the same at both Stations as well.
>
> The only real documentation difference of note was the use of Environmental Qualification Assessments (EQAs) at Bruce B and Environmental Qualification Dossiers (EQDs) at Bruce A, as the document containing the qualification basis information for EQ'd components at the respective stations. This historical difference in EQ documentation format doesn't pose any challenges and no action is proposed to address this difference.
>
> The advantage of utilizing a centralized corporate function to support Asset Management and Long Term Operations related activities was evident in interviews with Component and Programs Engineers, Nuclear Safety Analysis Support staff, and Corporate Asset Management Group staff.
>
> No adverse conditions or opportunities for improvement were identified in this FASA from the perspective of alignment or implementation of governance supporting Asset Management and Long Term Operation at Bruce A and Bruce B.

SA-MPR-2015-09 Inspection and Test Plans [183]

The scope of this assessment was to evaluate Pressure Boundary Inspection and Test Plans previously used by the Outage and Maintenance Services (OMS) Central Maintenance Weld Crew year to date in 2015. The objective was to determine the average time from field execution completion to Inspection and Test Plans (ITP) close-out and then measure that against the target timeline of < 60 days.

A total of 82 OMS Central Maintenance Mechanical Work orders with 79 Pressure Boundary ITPs used within the period of January 1, 2015 until July 1, 2015, were evaluated for this FASA.

The assessment concluded as follows:

"Upon a thorough review of all the information gathered it was determined that the average length of time from field execution completion to ITP completion and close-out is 57.1 days. Upon comparison to the target of < 60 days, we are currently exceeding the goal by 2.9 days on average, and therefore it is determined that no further action is required or recommended at this time. These timelines will continue to be monitored by the Section Manager of Inspection Services QC and by the PMC Field Engineering Section at least annually as a requirement of BP-PROC-00046 "Pressure Boundary Field Execution" section 4.15 "Verification of Procedure Compliance"."

SA-ERI-2015-11 System Performance Monitoring Plan (SPMP) Effectiveness [184]

This assessment was undertaken in order to evaluate System and Component Performance Monitoring Plans (CPMP/SPMP) for adherence to applicable governance, namely DPT-PE-00008, System and Component Monitoring Plans [72], and INPO 12-016 [185] on System Engineering Effectiveness. The assessment has determined that the existing Performance Monitoring Plans contain detailed requirements for either the Responsible Component Engineers or the Responsible System Engineers to use in the daily monitoring of equipment or system performance under their respective areas of responsibility which, in turn, forms the foundation of System Health Reporting.

In addition to the utilization of information contained within the Performance Monitoring Plans to identify degraded performance as noted above, other strengths were also identified during the assessment. These included the incorporation of performance goals and targets in the PMP, as well as, the utilization of degradation mechanisms and internal and external OPEX in order to further define functional failure modes.

A number of adverse conditions were identified, as follows:

"A general deficiency was observed with respect to the overall quality of the CPMP/SPMP's and the utilization of the PMP checklist to capture areas of weakness relative to procedural governance, DPT-PE-00008 [72] and INPO 12-016." SCR 28524656 was raised to address this finding and has since been completed.

"A deficiency was observed with respect to the inclusion of acceptance criteria and the revision frequency of the PMP's. In particular, several of the PMP's had not been updated within the 2 year frequency; they did not include acceptance criteria for the critical system parameters contained in the PMEL table and they did not include the identification of SPVs and critical spare strategies as prescribed by DPT-PE-00008 [72] and INPO 12-016." SCR 28524659 was raised to address this finding and is due for completion in August 2016.

"A deficiency was also observed with regards to the documentation of an equipment or system walkdown plan and whether acceptance criteria have been established for the critical parameters established in the walkdown plan. In particular, most of the PMP's evaluated did not have acceptance bands associated with the walkdown plans as required by DPT-PE-00008 [72] and INPO 12-016." SCR 28524663 was raised to address this finding and has since been completed.

"A deficiency was also noted with regards to the use and control of the PMP checklist. Currently, the PMP checklist is not included in the procedural governance, DPT-PE-00008 [72], and its use is, therefore, not controlled." SCR 28524665 was raised to address this finding and has since been completed.

"It is recommended that Plant Engineering at Bruce A review the results of this FASA and determine actions required to understand and address any extent of condition applicability to Bruce A Component or System Performance Monitoring Plans." SCR 28524895 was raised to address this finding and has since been completed.

7.2. Internal and External Audits and Reviews

The objective of the audit process as stated in BP-PROG-15.01 [186] is threefold:

- To assess the Management System and to determine if it is adequately established, implemented, and controlled;
- To confirm the effectiveness of the Management System in achieving the expected results and that risks are identified and managed; and
- To identify substandard conditions and enhancement opportunities.

The objective is achieved by providing a prescribed method for evaluating established requirements against plant documentation, field conditions and work practices. The process describes the activities associated with audit planning, conducting, reporting, and closing-out. The results of the independent assessments are documented and reported to the level of management having sufficient breadth of responsibility for resolving any identified problems (as stated in Section 5.14.2 of [187]).

This section contains information arising from audits related to ageing management. Internal audits are conducted by the Bruce Power Audit Department. External audits are conducted by independent third parties, excluding regulators.

7.2.1. Internal Audits

This assessment reviewed the relevant internal audits that were conducted in the five years since 2010. This includes the areas of:

- Plant Reliability Integration
- Inage Work Management
- Outage Work Management

- Plant Maintenance
- Chemistry Management

The audits reports were reviewed and the outstanding actions were checked to be complete or in progress. The results are shown in Table 9.

Table 9: Audits Relevant to SFR4 Conducted Since 2010

Audit Number	Title
Plant Reliability Integration	
AU-2010-00027	PHT Feeder MGMT
AU-2010-00037	GSA RV Field Repair Program
AU-2011-00007	RV field Repairs
AU-2011-00017	SST Scheduling and Completion
AU-2011-00018	Steam Generator Life Cycle Management
AU-2011-00025	Preventive Maintenance Deferral Process
AU-2011-00028	Performance and Condition Monitoring
AU-2012-00006	Equipment Reliability
AU-2012-00007	RV field Audits
AU-2013-00005	RV Field Repairs
AU-2014-00006	RV Program and Field Maintenance
AU-2014-00009	Compliance Evaluation to BP-PROC-00666 Component Categorization
AU-2014-00024	Compliance Evaluation: BP-PROC-00603 & BP-PROC-00789
AU-2015-00002	RV Program and Field Maintenance
Inage Work Management	
AU-2010-00022	H1/H2 Work Prioritization
AU-2012-00014	On-line Work Management
Outage Work Management	
AU-2010-00026	Forced Outage Management
AU-2013-00008	Outage Management

Audit Number	Title
Plant Maintenance	
AU-2010-00008	ISO 9001
AU-2010-00012	GSB Task Planning
AU-2011-00027	Foreign Material Exclusion
AU-2013-00006	Maintenance Program
Chemistry Management	
AU-2011-00024	Chemistry Management Program
AU-2011-00026	Outage Chemistry Program
AU-2014-00010	Control of System Chemistry

Four audits which were performed more recently and are more directly relevant to aging, are summarized below.

<u>AU-2013-00006, Maintenance Program</u> [188]

The objective of this audit was to evaluate whether BP-PROG-11.04 Plant Maintenance [49] is complete and fully implemented. This program governs the execution of required corrective maintenance when a critical SSC experiences an unplanned failure or when performance is seen, through Performance Monitoring, to have degraded.

The PROL in force at the time required that Bruce Power implement and maintain a maintenance program in accordance with CNSC regulatory document S-210 Maintenance Programs for Nuclear Power Plants [33]. BP-PROG-11.04 Plant Maintenance [49] is the program used to ensure compliance with S-210. This audit found that all major components and the majority of all the specific requirements of S-210 are covered in BP-PROG-11.04. However, BP-PROG-11.04 Plant Maintenance [49] is not fully complete and is not fully implemented since not all S-210 requirements are specifically addressed within the Program, although there are existing Bruce Power processes not cited in BP-PROG-11.04 [49] that satisfy the S-210 requirements [33].

Five adverse conditions and two opportunities for improvement were identified.

- Adverse Condition No. 1: BP-PROG-11.04 does not address all S-210 requirements
- Adverse Condition No. 2: Non-Maintenance Program processes that are relied upon by BP-PROG-11.04 to meet S-210 requirements are not identified as such
- Adverse Condition No. 3: BP-PROG-1 1.04 does not always specify the correct implementing process
- Adverse Condition No. 4: The Maintenance Program does not adequately cover Centre of Site Activities

- Adverse Condition No. 5: BP-PROG-1 1.04 does not always comply with BP-PROG-03.01 requirements
- Opportunity for Improvement No. 1: Clarification of BP-PROG-1 1.04 Information
- Opportunity for Improvement No. 2: Industrial Safety Reference

SCRs 28367179, 28367181, 28367185, 28367187, 28367192, 28367193 and 28367195 were raised to address these adverse conditions and opportunities for improvement. All assignments under these SCRs have been completed.

AU-2014-00010, Control of System Chemistry [189]

This audit evaluated the effectiveness of, and compliance to, DPT-CHM-00003 R006, Control of Chemistry [107]. DPT-CHM-00003 is relevant to aging management in controlling and moderating the rate of ageing degradation. This is accomplished by preventing inadvertent contact or intrusion of chemicals into plant systems that can result in chemistry excursions contributing to system degradation.

As documented in AU-2014-00010, DPT-CHM-00003 [107] requirements were generally found complete, established and implemented in accordance with its own requirements and the Bruce Power Management System. Six adverse conditions and three opportunities for improvement were identified, however no immediate negative consequences were found.

SCRs have been raised to address the adverse conditions and opportunities for improvement, as follows:

Adverse Conditions

- SCRs 28439133, 28439254, 28439134: Non-compliance to Control of Chemistry requirements. All assignments associated with these SCRs have been completed.
- SCRs 28439136, 28439262, 28439135: Chemistry Program Requirements are not adequate or complete. All assignments associated with these SCRs have been completed.
- SCRs 28439139, 28439264, 28439137: Audit (AU-2011-00024) and FASA (SA-CHM-2012-01) Corrective Actions found ineffective. All assignments associated with these SCRs have been completed, with the exception of one assignment of SCR 28439137 involving the revision of DPT-CHM-0006 which is due for completion in June 2016.
- SCRs 28439141, 28439265, 28439140: Control of Chemistry Program records not adequately controlled or maintained. All assignments associated with these SCRs have been completed.
- SCRs 28439143, 28439273, 28439142: Chemistry staff qualifications are not adequately established. Some assignments associated with these SCRs have been completed and most of them are scheduled for completion by the end of 2016. One assignment of SCR 28439273 is due for completion in June 2017.

- SCR 28439144: Control of Chemistry Program Non-Compliance to BPMS Procedural requirements. All assignments associated with this SCR have been completed.

Opportunities for Improvement:

- SCR 28439145: Control of Chemistry Program Description in BP-PROG-12.02 requires updating. All assignments associated with this SCR have been completed.
- SCR 28439146: Control of Chemistry SCA Trending expectations should be clearly documented and aligned. All assignments associated with this SCR have been completed.
- SCR 28439147: EPRI Strategic Water Chemistry Plans should be established. One assignment associated with this SCR is still open and due for completion in June 2016.

AU-2014-00009, Compliance Evaluation to BP-PROC-00666 Component Categorization [190]

The purpose of this audit was to validate that changes made to BP-PROC-00666 R002, Component Categorization, are being complied with, such that Engineering can validate procedural effectiveness and compliance with the recent revision, and to address any identified gaps in the procedure or its implementation that are not being met.

The overall conclusion of the Audit is that BP-PROC-00666, Component Categorization [52], is generally effective at achieving its purpose. However a lack of procedural compliance and deficiencies in the implementation have resulted in gaps between the expectations stated in the procedure and PassPort data for the Master Equipment List (MEL). There are four adverse conditions and one opportunity for improvement as identified below.

SCRs have been raised to address the adverse conditions and opportunity for improvement, as follows:

Adverse Conditions:

- SCR 28456027: AUDIT – Personnel do not always comply with BP-PROC-00666 R2. Personnel do not always comply with the requirements of BP-PROC-00666, and Equipment Information input into PassPort does not always comply with the requirements stated in BP-PROC-00666. All assignments associated with this SCR have been completed.
- SCR 28456029: AUDIT – Conflicting processes with BP-PROC-00666. There are conflicting processes that have resulted in non-compliances with the requirements of BP-PROC-00666. All assignments associated with this SCR have been completed.
- SCR 28456034: AUDIT – BP-PROC-00666 R2 not fully aligned with INPO AP-913. Definitions provided in BP-PROC-00666 Rev 2 do not completely align with INPO AP-913 Rev 4, in the areas of Power De-rates of less than 10%, Emergency Mitigation Equipment, and Maintenance Rule Requirements. All assignments associated with this SCR have been completed.

- SCR 28456045: AUDIT – BP-PROC-00666R2 errors, omissions, and misalignment. BP-PROC-00666 Revision 2 contains some errors, omissions, and misalignments with interfacing Controlled Documents, and information provided in PassPort. All assignments associated with this SCR have been completed.

Opportunity for Improvement:

- BP-PROC-00666 Section 4.0 indicates that Responsible System Engineers should perform periodic reviews on the Categorization of existing components. Staff indicated that aside from recent initiatives this does not occur.

AU-2014-00024, Compliance Evaluation: BP-PROC-00603 & BP-PROC-00789 [191]

The objective of this audit was to evaluate the compliance to BP-PROC-00603 R002 Preventive Maintenance Program "Just in Time" (JIT) Review Process [65] and BP-PROC-00789 R001 Maintenance Strategy [58]. Two adverse conditions were identified and SCRs have been raised to address them as follows:

Adverse Conditions:

- SCR 28473746 and 28473748: Non-compliances to BP-PROC-00603 R002 PM "JIT". BP-PROC-00603 is not followed as written. The JIT PM Process Review at Bruce A has evolved since June 2014. Bruce B has not yet implemented the JIT PM Process as envisioned in the procedure. The Bruce B JIT/7-26 meeting is solely for the purpose of engineering review and does not include the other stakeholders. All assignments associated with these SCRs have been completed.
- SCR 28473749: Maintenance Strategy process not fully established. The Asset Challenge Team (ACT) does not follow BP-PROC-00789 including the guidelines set out in its Appendix B. They rely instead on procedures that were developed by ACT during the Unit 1 & 2 project which are not controlled under Bruce Power's Management System and in most cases have not been approved by Bruce Power. Documented evidence of Review and Approval for PM Maintenance Strategies is not always clear. All assignments associated with this SCR have been completed.

7.2.2. External Audits and Reviews

In 2015 the World Association of Nuclear Operators (WANO) designated Bruce Power's obsolescence management process as a WANO Strength. Subsequently WANO produced a Good Practice Document with the Bruce B obsolescence management process as focus.

The IAEA Operational Safety Review Team (OSART) mission to Bruce B that took place from Nov. 30 to Dec. 17, 2015 noted that the plant has identified an opportunity to address obsolescence of technology:

> *The plant initiated preparation of a proactive obsolescence program which is currently being implemented. A part Readiness program was launched in Q4 2014 and combines Obsolescence Program, Critical Spare Parts Program and*

Catalogue Health Management. The first obsolescence solutions were put in place in 2015 with focus on the availability of safety-related spare parts. Clean-up of the spare parts catalogue was initiated in 2015 and first results will be available in 2016. Critical Spare Parts Program has objectives to further decrease unavailability of critical spare parts in 2016.

The OSART encouraged the plant to continue this proactive approach.

As a result of suggestions made by the OSART, the following Action Requests that relate to ageing management were opened and are in progress:

- AR 28552268 Assignment 02/03. An issue was identified under Conduct of Operations in that plant personnel do not consistently identify and report deficiencies in the area of Material Conditions, Equipment Labelling, Storage, and Operator Aids. The OSART team recommended that the plant should improve its standards for identifying and reporting deficiencies. An Operations Manager Expectation/Clarification has been distributed (OPMGR-EC-2016-00) to clarify the expectations for reporting deficiencies (GRP-OPS-00038) to include rust, damaged paint and oil sheens. These deficiencies are to be identified using Work Requests. An action to capture all deficiencies, including surface degradation in the Emergency Water & Power Supply Building (EWPSB) via Work Requests is in progress.
- AR 28554140. OSART suggested Bruce B consider improving equipment lists and attributes quality and completeness to support a comprehensive ageing management review of Long Term Operation (LTO). Completion of the action is due in 2018.
- A/R 28554146. OSART suggested that there is an opportunity to update ageing management programs for structures and components within the scope of LTO as part of ongoing reviews to ensure all aspects of LTO are considered in scope. Completion of the action is due in 2018.

7.3. Regulatory Evaluations and Reviews

After a licence is issued, the CNSC stringently evaluates compliance by the licensee on a regular basis. In addition to having a team of onsite inspectors, CNSC staff with specific technical expertise regularly visit plants to verify that licensees are meeting the regulatory requirements and licence conditions. Compliance activities include inspections and other oversight functions that verify a licensee’s activities are properly conducted, including planned Type I inspections (detailed audits), Type II inspections (routine inspections), assessments of information submitted by the licensee to demonstrate compliance, and other unplanned inspections in response to special circumstances or events.

Type I inspections are systematic, planned and documented processes to determine whether a licensee program, process or practice complies with regulatory requirements. Type II inspections are planned and documented activities to verify the results of licensee processes and not the processes themselves. They are typically routine inspections of specified equipment, facility material systems or of discrete records, products or outputs from licensee processes.

The CNSC carefully reviews any items of non-compliance and follows up to ensure all items are quickly corrected.

B-REP-00701-27MAY2013-051 [192] provides the results of an assessment of the status of the relevant CNSC inspections that were conducted since 2008. These inspections are identified in Table 3 of B-REP-00701-27MAY2013-051 [192]. This assessment concluded *inter alia* that Bruce Power should review the status of the recommendations in the following CNSC compliance inspections that apply to Bruce B:

- ID-BB-2008-13494-038: Structures, Systems, and Components Monitoring;
- IDB-2008-B-033-TI3082: Bruce B Maintenance Work Execution; and
- BRPD-AB-2012-011: Pressure Boundary Program Compliance at Bruce Power.

Subsequent follow-up at the time of preparing this Safety Factor report found the following:

- ID-BB-2008-13494-038: Structures, Systems, and Components Monitoring – this inspection was carried out at Bruce B, and the overall conclusion was that the management of Structures, Systems and Component monitoring for Bruce B meets requirements.
- IDB-2008-B-033-TI3082: Bruce B Maintenance Work – this inspection was carried out at Bruce B. The inspection report was provided to Bruce Power for information purposes only. No actions were placed on Bruce Power as a result of this inspection.
- BRPD-AB-2012-011: Pressure Boundary Program Compliance at Bruce – CNSC staff found that Bruce Power's implementation of the Pressure Boundary Program generally meets the requirements of the licences, licence condition handbooks and CSA-N285.0-08 Update No. 1. Three recommendations were raised as a result of this inspection. Bruce Power raised an Action Request to respond to these recommendations by December 2014.

In addition to the regulatory evaluations summarized in B-REP-00701-27MAY2013-051 [192], CNSC staff recently completed additional inspections relevant to ageing management. These are summarized below.

CNSC staff conducted a Type II inspection of Bruce Power's condition assessments in February 2014. In their compliance inspection report, BRPD-AB-2014-002, submitted under cover of NK29-CORR-00531-11783 [193], CNSC staff concluded that Bruce Power is aware of the condition of the systems at the Bruce facilities and has implemented measures to ensure that systems remain fit for service and meet regulatory requirements. Five action notices and three recommendations of relevance to Bruce B were raised as a result of this inspection, as follows (in Section 4):

Action Notice - BRPD-AB-2014-002-AN01:

> "In order to be compliant with NK29-CAR-33000-00001, section 7.1.1 and NK29-CAR-34330-00001, section 3.3.1, Bruce Power is requested to provide a status update of the PHT vibration issue, a description of the path forward to resolving the issue and to provide a description of the safety impact of the vibrations on the effected SSCs."

Action Notice - BRPD-AB-2014-002-AN02:

> "In order to be compliant with NK29-CAR-63720-00001, section 7.1.4, Bruce Power is requested to provide a status update on the status of the ultrasonic flow measurement system commissioning, a path forward to resolution of the issue, and assurances that the limits in the Safety Report continue to be met given this uncertainty."

Action Notice - BRPD-AB-2014-002-AN04:

> "In order to be compliant with BP-PROC-00781, section 4.6, Bruce Power is requested to demonstrate that the condition of the pre-stressing systems for the containment system is being monitored and that the condition is known."

Action Notice - BRPD-AB-2014-002-AN05:

> "In order to be compliant with the NK21-CAR-71300-00001, section 7.1, Bruce Power is requested to describe the risk of not having completed small projects in general which were assumed to be completed before refurbishment, or approximately 2014."

Action Notice - BRPD-AB-2014-002-AN06:

> "In order to be compliant with BP-PROC-00166 sections 4.4.24 and 4.4.25, Bruce Power shall review BP-PROC-00498 to ensure that the general procedure and process requirements are met. This review can be completed at the next revision of BP-PROC-00498."

Recommendation - BRPD-AB-2014-002-R01:

> "For any future condition assessments that are conducted, Bruce Power should ensure that all safety-related systems have reports produced."

Recommendation - BRPD-AB-2014-002-R02:

> "Bruce Power should ensure that any future condition assessment reports follow the established procedural requirements and the personnel adhere to the requirements."

CNSC raised Action Item 2014-07-4687 to track the actions arising from this inspection. Bruce Power responses to the Action Notices and Recommendations arising from this inspection are provided in NK29-CORR-00531-11921 [194] and follow-up correspondence. At time of writing of this report CNSC staff were reviewing additional information provided by Bruce Power via NK29-CORR-00531-12570 [195] on action notices BRPD-AB-2014-002-AN01 and BRPD-AB-2014-002-AN05 and considering a request for closure of Action Item 2014-07-4687 based on the information provided.

CNSC staff conducted a Type II inspection of Bruce Power's Reliability Program during September 2015. In their compliance inspection report, BRPD-AB-2015-008, submitted under cover of NK29-CORR-00531-12911 [196], CNSC staff concluded that based on the scope of this inspection, Bruce Power was in compliance with their licence and met the applicable regulatory requirements, with some non-compliances found with their procedures. As a result the following two action notices and nine recommendations have been raised:

Action Notice - BRPD-AB-2015-008-AN1:

"In order for Bruce Power to become compliant with BP-PROG-02.02 section 4.0.1 for the personnel that prepares and verifies [sic] the Annual Reliability Report and deferral reports, CNSC staff requests that Bruce Power provide a corrective action plan for incorporation of SAT based training including implementation milestone dates."

Action Notice - BRPD-AB-2015-008-AN2:

"In order for Bruce Power to become compliant with BP-PROG-11.01 sub-section 3.1.22 item #2, CNSC staff request Bruce Power to develop and implement a corrective action plan to come up with a method of ranking for the new updated S-294 PSA systems on the SIS list using combination of importance measures."

Recommendation - BRPD-AB-2015-008-R1:

"Bruce Power to include the procedures and work instructions with the mapping of the RD-98 requirements in BP-PROG-11.01.

Recommendation - BRPD-AB-2015-008-R2:

"That Bruce Power updating [sic] the unavailability model reports once the S-294 Reliability models are operationalized for inclusion into the reliability program."

Recommendation - BRPD-AB-2015-008-R3:

"That Bruce Power establish and document the process for dispositioning requests for adding new failure modes, discovered yearly from operation, into the PSA and Unavailability models."

Recommendation - BRPD-AB-2015-008-R4:

"That Bruce Power aligns the efforts of the responsible system engineers and the reliability group for identification of failure modes through FMEA."

Recommendation - BRPD-AB-2015-008-R5:

"Bruce Power to include a field for capturing CCFs in the NuREP database."

Recommendation - BRPD-AB-2015-008-R6:

"That Bruce Power ensure that all procedures and task books related to the Reliability program are up to date."

Recommendation - BRPD-AB-2015-008-R7:

"Bruce Power to clearly document the process for data collection of standby generator attempts to start or run, the details on processing overlapping faults and the method for instantaneous unavailability calculations."

Recommendation - BRPD-AB-2015-008-R8:

"That Bruce Power provide regular updates to CNSC on the implementation of the plan for updating SIS list."

Recommendation - BRPD-AB-2015-008-R9:

> "That Bruce Power makes reference to the document that the failure criteria is [sic] derived from in the unavailability model report."

CNSC raised Action Item 2015-07-7231 to track the actions arising from this inspection. Bruce Power's responses to the Action Notices and Recommendations arising from this inspection are provided in NK29-CORR-00531-13044 [197]. In this response Bruce Power provides a schedule of milestones to fully address action item BRPD-AB-2015-008-AN1 and AN2 by November 2016 and April 1, 2018 respectively and request that Action Item 2015-07-7231 therefore be closed. Bruce Power also indicated that it will consider recommendations BRPD-AB-2015-00B-R1 to BRPD-AB-2015-00B-R9 during the related documentation revision cycles.

7.4. Performance Indicators

Performance indicators are defined as data that are sensitive to and/or signals changes in the performance of systems, components, or programs.

For components, specific performance indicators on aging and obsolescence are monitored. Other performance indicators may indicate ageing-related changes in the performance of a system or component, such as:

- Functional Failures (number of functional failures and outstanding functional failure corrective actions)
- Maintenance backlogs (online deficient maintenance backlog, shutdown deficient maintenance backlog, online corrective maintenance backlog, shutdown corrective maintenance backlog and predefines – total of late and deferred)
- Equipment Reliability Clock Resets.

Additional performance indicators for systems, such as operational challenges (i.e., open Technical Operability Evaluation items and CNSC REGDOC-3.1.1 reportable events), may also indicate ageing related issues. Figure 3 is an extract from a recent SHR for the Standby Generators that serves as an example of how these performance indicators are used. The calculation of aggregate scores is based on weighting and normalization of the various indicators. The Health History shows change in system health over time based on the performance indicators.

Health History

	Status	-1 Status	-2 Status	-3 Status
Overall		WHITE	YELLOW	YELLOW
Fuel		WHITE	WHITE	RED
SG 5		GREEN	YELLOW	YELLOW
SG 6		YELLOW	YELLOW	YELLOW
SG 7		YELLOW	YELLOW	YELLOW
SG 8		WHITE	WHITE	WHITE

Station Summary

Performance Indicator Data	**Fuel**	**SG 5**	**SG 6**	**SG 7**	**SG 8**
Functional Failures					
Number of Functional Failures	1	1	6	1	1
Outstanding Functional Failure Corrective Actions	0	0	1	1	1
Maintenance Backlog					
ODMB (On-Line Deficient Maintenance Backlog)	10	8	17	18	14
SDMB (Shutdown Deficient Maintenance Backlog)	0	0	1	1	0
OCMB (On-Line Corrective Maintenance Backlog)	4	0	3	6	2
SCMB (Shutdown Corrective Maintenance Backlog)	0	0	0	0	0
Predefines - Total of Late and Deferred	0	8	14	17	8
Operational Challenges					
Open TOE Items	0	0	0	0	0
S99 Reportable Events	0	0	0	0	0
Engineering					
TMOD > 6 Months	6	0	0	0	0
Temporary Configuration Change Backlog > 90 Days	6	0	0	1	0
Modification Backlog	2	1	4	1	2
Operator Challenges					
Operator Workarounds	1	0	0	0	0
Operator Burdens	0	0	0	0	1
Additional and Specific Indicators					
Active OPMs	5	0	0	1	1
ER Clock Resets	0	0	1	0	0
Forced Outages	0	0	3	0	0
Number of H WO's	0	0	3	0	0
Number of SCRs	26	10	3	2	0
Open ESR flagged WOs	0	0	0	0	0
Open SHR flagged WO's	0	0	0	0	0
Open SHY flagged WO's	0	0	1	0	1
Plant IQ	0	1	1	2	2
Running Failures	0	0	1	0	0
SST Failures	2	0	2	0	0
Start Failures	0	0	2	0	0
WANO Emergency AC Power - 3 Year Monthly Rolling Average	0	77	3150	3442	122
Total:	73.67	88.34	55	80.67	79.67

Figure 3: Example of the Use of Performance Indicators in System Health Reports

For Bruce Power engineering programs, performance indicators are grouped under Performance Cornerstones. Program Health Reports for these programs include data on a number of mandatory cornerstones including:

- Personnel Cornerstones (program owner qualification and experience, backup program owner qualification and experience, industry participation)
- Infrastructure Cornerstones (long range plan, open program enhancement action requests / SCRs, program infrastructure deficiency notifications / SCRs)
- Implementation Cornerstones (self-assessment, OPEX implementation, program implementation notifications)
- Equipment Cornerstones (critical component failure, adverse failure trend, life cycle management plan).

The following engineering programs are relevant to ageing management:

- Buried Piping
- Flow Accelerated Corrosion
- Periodic Inspection
- Pipe Support Inspection
- Preventive Maintenance
- Predictive Maintenance
- Strategic / Critical Spares.

8. Summary and Conclusions

The overall objectives of the Bruce B PSR are to conduct a review of Bruce B against modern codes and standards and international safety expectations, and to provide input to a practicable set of improvements to be conducted during the MCR in Units 5 to 8, as well as U0B, and during asset management activities to support ongoing operation of all four units, that will enhance safety to support long term operation. The specific 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. This specific objective has been met by the completion of the review tasks specific to ageing.

Strengths identified during this review are 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.

- 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.

Table 10 summarizes the key issues arising from the Integrated Safety Review of Safety Factor 4.

Table 10: Key Issues

Issue Number	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
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

Overall, ageing management at Bruce Power meets the requirements of the Safety Factor related to ageing. The review indicates that the current and planned implementations of the programs related to ageing are sufficient to support continued operation of Bruce B.

9. References

[1] NK21-CORR-00531-12136/NK29-CORR-00531-12546/E-DOC 4723908, Nuclear Power Reactor Operating Licence, Bruce Nuclear Generating Stations A and B, PROL 18.00/2020, Canadian Nuclear Safety Commission, May 27, 2015.

[2] NK21-CORR-00531-12135/NK29-CORR-00531-12545/E-DOC 4659316, Licence Conditions Handbook, LCH-BNGS-R000, Bruce Nuclear Generating Station A and Bruce Nuclear Generating Station B Nuclear Reactor Operating Licence, PROL 18.00/2020 (Effective: June 1, 2015), Canadian Nuclear Safety Commission, May 27, 2015.

[3] CNSC REGDOC-2.3.3, Operating Performance: Periodic Safety Reviews, CNSC, April 2015.

[4] BP-PROC-01024-R000, Periodic Safety Reviews, Bruce Power, November 7, 2015.

[5] NK29-CORR-00531-12932, Bruce B Periodic Safety Review Basis Document, Bruce Power Letter, F. Saunders to K. Lafrenière, January 25, 2016.

[6] NK21-CORR-00531-11005/NK29-CORR-00531-11397, Submission of Safety Basis Report, Bruce Power Letter, F. Saunders to R. Lojk, December 30, 2013.

[7] NK21-CORR-00531-00514, Bruce A: CNSC Approval to Restart Units 3 and 4 and Application to Amend PROL 15.01/2003, Bruce Power Letter, F. Saunders to J.H.M. Douglas, November 16, 2001.

[8] NK21-CORR-00531-04636, Bruce A Units 1 and 2 Return to Service: Systematic Review of Safety – Basis, Bruce Power Letter, F. Saunders to D.A. Desjardins, December 22, 2006.

[9] NK21-CORR-00531-04059, Bruce A Refurbishment for Life Extension – Systematic Review of Safety: Plant Design, Bruce Power Letter, F. Saunders to P. Webster, March 30, 2006.

[10] NK21-CORR-00531-04339, Bruce A Units 1 and 2 Return to Service – Systematic Review of Safety, Bruce Power Letter, F. Saunders to P. Webster, July 31, 2006.

[11] NK21-CORR-00531-05749, Bruce A Refurbishment for Life Extension and Continued Operation of Units 3 and 4 – Integrated Safety Review Basis, F. Saunders to P. Elder, February 29, 2008.

[12] NK21-CORR-00531-05976, Bruce A Units 3 and 4 Refurbishment for Life Extension and Continued Operations: ISR Safety Factor Reports, Bruce Power Letter, F. Saunders to P. Elder, June 2, 2008.

[13] NK21-CORR-00531-06596, Bruce A Units 3 and 4 Refurbishment for Life Extension and Continued Operation: ISR Safety Factor Reports 1, 2, 3 and 4, Bruce Power Letter, F. Saunders to K. Lafrenière, December 18, 2008.

[14] NK21-CORR-00531-06076, Bruce A Units 3 and 4 Refurbishment for Life Extension and Continued Operation: ISR Safety Factor Reports 5, 6, and 7, Bruce Power Letter, F. Saunders to P. Elder, July 22, 2008.

[15] NK21-REP-03600-00025-R001, Bruce NGS A Units 3 and 4 Global Assessment Report and Integrated Implementation Plan, May 29, 2009.

[16] NK21-CORR-00531-11617, Integrated Safety Review for Bruce A, Bruce Power Letter, F. Saunders to K. Lafrenière, including enclosure K-421231-00010-R00, Candesco Report, October 27, 2014.

[17] NK21-CORR-00531-12269, Bruce A Integrated Safety Review – Safety Factor Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, August 27, 2015.

[18] NK21-CORR-00531-10576/NK29-CORR-00531-10975, Application Requirements for Renewal of Power Reactor Operating Licences for Bruce Nuclear Generating Stations A and B, Bruce Power Letter, F. Saunders to R. Lojk, July 17, 2013.

[19] NK29-CORR-00531-11252, Application for the Renewal of the Power Reactor Operating Licence for Bruce Nuclear Generating Station B, Letter, F. Saunders to M. Leblanc, October 31, 2013.

[20] NK21-CORR-00531-11711/NK29-CORR-00531-12101, Bruce A and Bruce B Licence Renewal – Supplemental Update, F. Saunders to M. Leblanc, November 27, 2014.

[21] NK21-CORR-00531-11715/NK29-CORR-00531-12105, Bruce Power: Requests and Supplemental Information for Licence Renewal, F. Saunders to M. Leblanc, November 28, 2014.

[22] Nuclear Safety and Control Act, 1997, c. 9, N-28.3, Assented to March 20, 1997.

[23] CNSC REGDOC-3.1.1, Reporting Requirements for Nuclear Power Plants, CNSC, May 2014.

[24] CNSC RD/GD-210, Maintenance Programs for Nuclear Power Plants, CNSC, November 2012.

[25] CNSC RD/GD-98, Reliability Programs for Nuclear Power Plants, June 2012.

[26] CNSC REGDOC-2.6.3, Fitness for Service: Aging Management, March 2014.

[27] CAN/CSA-N285.4-14, Periodic Inspection of CANDU Nuclear Power Plant Components, May 2014.

[28] CAN/CSA-N285.5-13, Periodic Inspection of CANDU Nuclear Power Plant Containment Components, July 2013.

[29] CAN/CSA-N287.7-08, In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, 2008, Re-affirmed 2013 (including Update 1, Issued September 2010).

[30] CAN/CSA-N286-12, Management System Requirements for Nuclear Power Plants, June 2012.

[31] CAN/CSA-N290.13-05, Environmental Qualification of Equipment for CANDU Nuclear Power Plants, October 2009.

[32] CNSC S-99, Reporting Requirements for Operating Nuclear Power Plants, March 2003.

[33] CNSC S-210, Maintenance Programs for Nuclear Power Plants, July 2007.

[34] NK21-CORR-00531-05204, Bruce A Units 1 and 2 Return to Service – Systematic Review of Safety: Category 2 Issues, Bruce Power Letter, F. Saunders to P. Elder, August 29, 2007.

[35] CNSC S-98-R001, Reliability Programs for Nuclear Power Plants, July 2005.

[36] BP-PROG-11.01-R005, Equipment Reliability, Bruce Power, December 16, 2015.

[37] CNSC RD-334, Aging Management for Nuclear Power Plants, June 2011.

[38] NK21-CORR-00531-11763/NK29-CORR-00531-12158, Transition Plan for REGDOC-2.6.3 Implementation, Bruce Power Letter, F. Saunders to K. Lafrenière, December 12, 2014.

[39] CAN/CSA-N285.4-09, Periodic Inspection of CANDU Nuclear Power Plant Components, 2009.

[40] NK21-CORR-00531-11494/NK29-CORR-00531-11881, Response to Bruce Power letter, CSA N286-12 “Management system requirements for nuclear facilities”, Action Item 1307-4697, CNSC Letter, K. Lafrenière to F. Saunders, July 24, 2014.

[41] NK21-CORR-00531-11189/NK29-CORR-00531-11593, Action Item 1307-4697: CSA N286-12 Management System Requirements for Nuclear Facilities, Bruce Power Letter, F. Saunders to K. Lafrenière, May 16, 2014.

[42] NK21-CORR-00531-11563/NK29-CORR-00531-11946, CSA N286-12 - Management System Requirements for Nuclear Facilities, Action Item 1307-4697, Bruce Power Letter, F. Saunders to K. Lafrenière, September 9, 2014.

[43] NK21-CORR-00531-12570/NK29-CORR-00531-12996, Action Item 1307-4697: CSA N286-12 - Management Systems Requirements for Nuclear Facilities, Bruce Power Letter, F. Saunders to K. Lafrenière, January 29, 2016.

[44] CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants, May 2014.

[45] CAN/CSA-N285.8-15, Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors, July 2015.

[46] CAN/CSA-N291-15, Requirements for Safety-Related Structures for Nuclear Power Plants, November 2015.

[47] CAN/CSA-N287.1-14, General Requirements for Concrete Containment Structures for Nuclear Power Plants, February 2014.

[48] IAEA SSG-25, Periodic Safety Review of Nuclear Power Plants, IAEA, March 2013.

[49] BP-PROG-11.04-R006, Plant Maintenance, Bruce Power, November 27, 2013.

[50] BP-MSM-1-R012, Management System Manual, Bruce Power, June 23, 2014.

[51] BP-PROC-00778-R002, Scoping and Identification of Critical SSCs, Bruce Power, September 3, 2015.

[52] BP-PROC-00666-R004, Component Categorization, Bruce Power, October 1, 2015.

[53] DPT-RS-00012-R001, Systems Important to Safety (SIS) Decision Methodology, Bruce Power, September 24, 2013.

[54] BP-PROC-00779-R001, Continuing Equipment Reliability Improvement, Bruce Power, September 14, 2015.

[55] BP-PROC-00532-R002, Critical and Strategic Spares, Bruce Power, March 5, 2014.

[56] BP-PROC-00534-R003, Technical Basis Assessment, Bruce Power, September 2015.

[57] BP-PROC-00539-R016, Design Change Package, Bruce Power, June 23, 2015.

[58] BP-PROC-00789-R001: Maintenance Strategy, Bruce Power, April 23, 2014.

[59] BP-PROC-00780-R002, Preventive Maintenance Implementation, Bruce Power, September 11, 2015.

[60] BP-PROC-00284-R004, Predictive Maintenance, Bruce Power, October 13, 2015.

[61] BP-PROC-00456-R009, Preventative Maintenance (PM) WO Deferral Process, Bruce Power, January 30, 2015

[62] BP-PROC-00457-R007, Development and Approval of Predefined, Bruce Power, January 23, 2013.

[63] BP-PROC-00501-R005, Integrated Preventive Maintenance Program, Bruce Power, May 27, 2014.

[64] BP-PROC-00599-R000, Engineering Guidance for Preventative Maintenance Program Support, August 20, 2008.

[65] BP-PROC-00603-R003, Preventive Maintenance Just-in-Time [JIT] Review Process, Bruce Power, May 11, 2015.

[66] SEC-MSS-00004-R000, Proactive Maintenance Process, Bruce Power, June 14, 2006.

[67] BP-PROC-00781-R003, Performance Monitoring, Bruce Power, September 11, 2015.

[68] BP-PROC-00361-R002, In-Service Testing and Inspection to Satisfy CAN/CSA-N287.7-08 Requirements, Bruce Power, March 13, 2015.

[69] BP-PROC-00387-R001, Plant Inspection, Bruce Power July 29, 2015.

[70] BP-PROC-00893-R000, Fuel and Fuel Channel Program, Bruce Power, December 10, 2012.

[71] DPT-PE-00005-R000, Performance Requirements for Contamination Exhaust Control Filters, Bruce Power, February 23, 2005.

[72] DPT-PE-00008-R006, System/Component Performance Monitoring Plan, Bruce Power, September 11, 2015.

[73] DPT-PE-00009-R002, System and Component Performance Monitoring Walkdowns, October 2015.

[74] DPT-PE-00010-R006, System Health Reporting, Bruce Power, August 27, 2013.

[75] DPT-PE-00011-R003, Component Health Reporting, Bruce Power, August 23, 2013.

[76] BP-PROC-00782-R001, Equipment Reliability Problem Identification and Resolution, Bruce Power, October 14, 2014.

[77] BP-PROC-00496-R005, Trouble Shooting Plant Equipment, Bruce Power, April 2015.

[78] DIV-ENG-00004-R008, Engineering Evaluations, Bruce Power, April 11, 2014.

[79] BP-PROC-00783-R001, Long Term Planning and Life Cycle Management, Bruce Power, September 25, 2013.

[80] BP-PROC-00400-R002, Life Cycle Management of Critical SSCs, Bruce Power, July 5, 2013.

[81] BP-PROC-00533-R002, Obsolescence Management, Bruce Power, October 28, 2014.

[82] AP-913-R004, Equipment Reliability Process Description, Institute of Nuclear Power Operations, November 2001.

[83] BP-PROC-00169-R002, Safety-Related System List, Bruce Power, September 28, 2007.

[84] CAN/CSA-N290.15-10, Requirements for the Safe Operating Envelope of Nuclear Power Plants, August 2010.

[85] DPT-RS-00015-R000, Safe Operating Envelope Gap Assessment, Bruce Power, May 31, 2011.

[86] BP-PROC-00060-R028, Station Condition Record Process, Bruce Power, November 5, 2015.

[87] BP-PROC-00019-R010, Action Tracking, Bruce Power, May 12, 2015.

[88] FORM-14071-R000, Equipment Failure Checklist, Bruce Power.

[89] BP-PROC-00559-R004, Station Plant Health Committee, Bruce Power, November 21, 2013.

[90] BP-PROC-00695-R003, Maintenance Program and Activities, Bruce Power, June 27, 2014.

[91] BP-PROC-00696-R002, Maintenance Organization, Bruce Power, December 2, 2013.

[92] BP-PROC-00699-R003, Maintenance Work, Bruce Power, December 5, 2013.

[93] BP-PROG-12.01-R007, Conduct of Plant Operations, Bruce Power, August 13, 2013.

[94] GRP-OPS-00038-R009, Bruce A and B Operations Standards and Expectations, Bruce Power, October 31, 2013.

[95] BP-OPP-00001-R019, Operating Policies and Principles – Bruce B, Bruce Power, July 14, 2015.

[96] BP-PROC-00734-R005, Plant Status Control, Bruce Power, April 22, 2014.

[97] GRP-OPS-00026-R002, Logging, Bruce Power, May 25, 2015.

[98] GRP-OPS-00047-R007, Operator Routines and Inspections – Bruce A and Bruce B, December 10, 2013.

[99] GRP-OPS-00001-R006, Operating Memos: Bruce A and Bruce B, Bruce Power, March 6, 2015.

[100] BP-PROC-00059-R022, Event Response and Reporting, Bruce Power, December 11, 2014.

[101] BP-PROC-00519-R009, Apparent Cause Evaluation (ACE), Bruce Power, August 12, 2014.

[102] BP-PROC-00518-R007, Root Cause Investigation, Bruce Power, July 15, 2014.

[103] BP-PROC-00014-R009, Technical Operability Evaluation, Bruce Power, October 23, 2014.

[104] GRP-OPS-00030-R006, Operational Decision Making. Bruce Power November 21, 2013.

[105] DPT-NSAS-00016-R000, Integrated Aging Management for Safety Assessment, Bruce Power, June 21, 2013.

[106] BP-PROG-12.02-R006, Chemistry Management, Bruce Power, June 8, 2015.

[107] DPT-CHM-00003-R008, Control of Chemistry, Bruce Power, July 16, 2015.

[108] DPT-CHM-00007-R008, Performance Monitoring, Bruce Power, September 2, 2010.

[109] DPT-CHM-00008-R006, Outage Chemistry Program, Bruce Power, June 8, 2015.

[110] BP-PROG-10.01-R009, Plant Design Basis Management, Bruce Power, December 4, 2014.

[111] BP-PROG-10.02-R010, Engineering Change Control, Bruce Power, November 12, 2014.

[112] BP-PROG-10.03-R006, Configuration Management, Bruce Power, February 5, 2015.

[113] BP-PROC-00923-R000, Pipe Wall Thinning – FAC, Bruce Power, January 16, 2014.

[114] BP-PROC-00334-R003, Periodic Inspection, Bruce Power, May 2, 2014.

[115] BP-PROC-00825-R001, Buried Piping Inspection Program, Bruce Power, March 24, 2015.

[116] NK29-PIP-03641.2-00001-R004, Bruce B Periodic Inspection Plan for Unit 5, March 4 2015.

[117] NK29-PIP-03641.2-00002-R004, Bruce B Periodic Inspection Plan for Unit 6, March 2, 2015.

[118] NK29-PIP-03641.2-00003-R004, Bruce B Periodic Inspection Plan for Unit 7, March 3 2015.

[119] NK29-PIP-03641.2-00004-R004, Bruce B Periodic Inspection Plan for Unit 8, March 4 2015.

[120] B-PLAN-31100-00001-R005, Fuel Channel Life Cycle Management Plan (FCLCMP), October 18, 2012.

[121] B-PIP-31100-00001-R003, Fuel Channel Periodic Inspection Program, April 2012.

[122] B-LCM-33126-00001-R000, PHT Feeder Piping LCMP, September 29, 2014.

[123] B-PLAN-33110-00001-R004, Steam Generator and Preheater Life Cycle Management Plan, February 17, 2011.

[124] NK29-PIP-03642-00001-R002, Bruce NGS B Periodic Inspection Plan for Unit 0 and Units 5 to 8 Containment Components, October 1, 2015.

[125] BP-PROC-00323-R001, Predictive Maintenance Lubrication Analysis, Bruce Power, December 4, 2006.

[126] BP-PROC-00762-R000, Predictive Maintenance Ultrasound Inspection Program, Bruce Power, May 25, 2010.

[127] BP-PROC-00768-R001, Predictive Maintenance Infrared Thermography Program, Bruce Power, November 24, 2014.

[128] SEC-RE-00009-R001, Predictive Maintenance Vibration Monitoring Program, May 8, 2008.

[129] SEC-RE-00016-R000, Predictive Maintenance Motor Testing Program, May 25, 2009.

[130] BP-PROG-10.02-R010, Engineering Change Control, Bruce Power, December 2014.

[131] BP-PROG-11.02-R006, On-Line Work Management Program, Bruce Power, October 23, 2012.

[132] BP-PROC-00329-R016, On-Line Work Management Process, Bruce Power, September 8, 2015.

[133] BP-PROC-00328-R015, Work Prioritization and Approval, Bruce Power, July 31, 2015.

[134] NK21-CORR-00531-10769/NK29-CORR-00531-11151, Periodic Safety Review: Status of Maintenance Backlogs, Composite Safety Profile Methodology and Bruce Power Periodic Safety Review Program Document, Bruce Power Letter, F. Saunders to R. Lojk, October 1, 2013

[135] NK21-CORR-00531-11698/NK29-CORR-00531-12088, Bruce A and B Quarterly Field Inspection Report for Q2 of 2014-15 BRPD-AB-2014-011, CNSC Letter, K. Lafrenière to F. Saunders, November 4, 2014.

[136] NK21-CORR-00531-10818/NK29-CORR-00531-11190, Bruce A and B: Maintenance Planning and Scheduling Inspection BRPD-AB-2013-004, Action Item 1307-4113, CNSC Letter, R. Lojk to F. Saunders, September 25, 2013.

[137] NK21-CORR-00531-12503/NK29-CORR-00531-12928, Action Item 1307-4113: Bruce A and B Maintenance Planning and Scheduling Inspection BRPD-AB-2013-004, CNSC Letter, K. Lafrenière to F. Saunders, November 23, 2015.

[138] BP-PROC-00638-R012, Temporary Configuration Change Management, Bruce Power, May 17, 2014.

[139] B-REP-00701-23SEP2013-057, Assessment of System and Component Health Reports, Bruce Power, September 2013.

[140] N-REP-03611-0381169/eDoc 3413831, Application of the CNSC Risk-Informed Decision Making Process to Category 3 CANDU Safety Issues, Canadian Nuclear Safety Commission, August 2009.

[141] NK21-CORR-00531-10027/NK29-CORR-00531-10447, CANDU Category III Safety Issues: Request for Reclassification of CI1, GL3 and PF19, Bruce Power Letter, F. Saunders to G. Rzentkowski, December 5, 2012.

[142] NK21-CORR-00531-10437/NK29-CORR-00531-10839, Reply to Bruce Power Request for Reclassification of Category 3 CANDU Safety Issue PF 19: Impact of Aging on Safe Plant Operation, CNSC Letter, G. Rzentkowski to F. Saunders, April 18, 2013.

[143] NK21-CORR-00531-10895/NK29-CORR-00531-11238, Reply to Request for Re-Categorization of CANDU Safety Issue CI-1: Fuel Channel Integrity and Effect on Core Internals, CNSC Letter, G. Rzentkowski to F. Saunders, October 22, 2013.

[144] NK21-CORR-00531-11280/NK29-CORR-00531-11688, Reply to Request for Re-categorization of CANDU Safety Issue GL 3: Aging of Equipment and Structures, CNSC Letter, G. Rzentkowski to F. Saunders, April 16, 2014.

[145] NK29-PIP-20000-00001-R000, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures, Bruce Power, September 2014.

[146] BP-PROG-03.01-R016, Document Management, Bruce Power, August 31, 2015.

[147] BP-PROC-00068-R023, Controlled Document Life Cycle Management, Bruce Power, December 11, 2015.

[148] BP-PROC-00098-R015, Records Management, Bruce Power, November 27, 2014.

[149] BP-PROC-00972-R000, Records Retrieval and Secure Storage, Bruce Power, November 21, 2014.

[150] BP-PROG-03.02-R004, Information Technology, Bruce Power, June 2, 2014.

[151] BP-PROC-00124-R005, IT Solutions – Enterprise Asset Management Support, Bruce Power, March 14, 2011.

[152] IAEA Safety Standards, Safety Guide No. NS-G-2.12, Ageing Management for Nuclear Power Plants, January 2009.

[153] NK21-CORR-00531-10278/NK29-CORR-00531-10862, Reply to Re-categorization Request of CANDU Safety Issue GL3: Ageing of Equipment and Structures, CNSC Letter, G. Rzentkowski to F. Saunders, February 18, 2013.

[154] NK21-CORR-00531-10336/NK29-CORR-00531-10745, CANDU Safety Issue GL3: Aqeing of Equipment and Structures, Bruce Power Letter, F. Saunders to G. Rzentkowski, April 4, 2013.

[155] FORM-10700-R021, Design Scoping Checklist, Bruce Power.

[156] BP-PROC-00786-R003, Margin Management, Bruce Power, August 5, 2014.

[157] B-REP-31100-00003-R003, Fuel Channel Condition Assessment, Bruce Power, December 2011.

[158] CSA-N285.8-05, Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors, June 2005.

[159] CAN/CSA-N285.8-10, Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors, Update No. 2 June 2013, 2010.

[160] NK21-CORR-00531-12487/NK29-CORR-00531-12902, Bruce A and B: Compliance Plan to CSA N285.8-10, Bruce Power Letter, F. Saunders to K. Lafrenière, April 27, 2016.

[161] NK21-CORR-00531-12856/NK29-CORR-00531-13312, Bruce A and B: Compliance Plan to CSA N285.8-10, Bruce Power Letter, K. Lafrenière to F. Saunders, May 27, 2016.

[162] B-REP-31100-00010-R002, Evaluation Process of Pressure Tube Fitness-for-Service Using CSA-N285.8, August 2014.

[163] BP-PROC-00731-R000, PHT Feeder Piping Life Cycle Program, Bruce Power, December 14, 2009.

[164] CSA-N285.4-05, Periodic Inspection of CANDU Nuclear Power Plant Components, June 2005.

[165] DPT-ENG-00019-R005, Disposition of PIP/In-Service Inspection Results, Bruce Power, September 2, 2015.

[166] BP-PROC-00267-R000, Management of Steam Generator and Preheater Tube Integrity, Bruce Power, November 30, 2004.

[167] Bruce Power Life Cycle Management Plan Summaries, email from M. Burton, Bruce Power, to J. Stevenson, CNSC, February 29, 2016.

[168] NK21-CORR-00531-10943/NK29-CORR-00531-11325, Safety Analysis in Support of Operation of Bruce A and Bruce B to 2019, Bruce Power Letter, F. Saunders to R. Lojk, December 12, 2013.

[169] BP-PROC-00383-R000, Performance and Condition Assessment, Bruce Power, December 15, 2006.

[170] BP-PROG-05.01-R014, Supply Chain, Bruce Power, February 4, 2015.

[171] BP-PROC-00262-R008, Warehouse Operations, Bruce Power, August 27, 2014.

[172] BP-PROC-00999-R000, Selection of Shelf Life Requirements, Bruce Power, March 13, 2015.

[173] DPT-MM-00007-R011, Control of Item Shelf Life Management, January 12, 2016.

[174] BP-PROC-00135-R005, Station Rework Program, Bruce Power, October 4, 2012.

[175] IAEA Safety Reports Series No. 82, Ageing Management for Nuclear Power Plants: International Generic Ageing Lessons Learned (IGALL), IAEA, Vienna, 2015.

[176] SA-ERI-2012-04, Assessing Life Cycle Management Plan Effectiveness, June 26, 2012.

[177] SA-ERI-2013-03, System and Component Performance Monitoring Program Compliance, December 20, 2013.

[178] SA-ERI-2014-01, Review of Data Needs to Assess SSC Aging, October 17, 2014.

[179] SA-ERI-2014-07, Quality of System Health Reporting, August 21, 2014.

[180] SA-MPR-2014-08, SECNMMM Equipment Capability, August 2014.

[181] SA-ERI-2015-02, Use of Condition Based Maintenance for Scheduling Decision, November 29, 2015.

[182] SA-ERI-2015-04, Alignment of ER Governance Implementation at Bruce A and Bruce B, December 14, 2015

[183] SA-MPR-2015-09, Inspection and Test Plans, September 2015.

[184] SA-ERI-2015-11, System Performance Monitoring Plan (SPMP) Effectiveness, October 2015.

[185] INPO 12-016, System Engineering Effectiveness, Institute of Nuclear Power Operations, December 2012.

[186] BP-PROG-15.01-R004, Nuclear Oversight Management, Bruce Power, December 18, 2013.

[187] CAN/CSA N286-05, Management System Requirements for Nuclear Power Plants, CSA, February 2005, (Update No.2, 2010), R2010.

[188] AU-2013-00006, Maintenance Program, April 2013.

[189] AU-2014-00010, Control of System Chemistry, May 1, 2014.

[190] AU-2014-00009, Compliance Evaluation to BP-PROC-00666 Component Categorization, September 2014.

[191] AU-2014-00024, Compliance Evaluation: BP-PROC-00603 & BP-PROC-00789, November 2014.

[192] B-REP-00701-27MAY2013-051, Assessment of the Ageing Management Program, Bruce Power, May 2013.

[193] NK21-CORR-00531-11380/NK29-CORR-00531-11783, Action Item 2014-07-4687: CNSC Type II Inspection - Condition Assessment Inspection - BRPD-AB-2014-002, CNSC Letter, K. Lafrenière to F. Saunders, June 3, 2014.

[194] NK21-CORR-00531-11534/NK29-CORR-00531-11921, Action Item 2014-07-4687: Bruce Power Responses to CNSC Type II Inspection - Condition Assessment Inspection - BRPD-AB-2014-002, Bruce Power Letter, F. Saunders to K. Lafrenière, September 9, 2014.

[195] NK21-CORR-00531-12206/NK29-CORR-00531-12570, 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.

[196] NK21-CORR-00531-12495/NK29-CORR-00531-12911, Action Item 2015-07-7231: CNSC Type II Compliance Inspection Report: BRPD-AB-2015-008 Implementation of the Bruce Power Reliability Program, CNSC Letter, K. Lafrenière to F. Saunders, November 25, 2015.

[197] NK21-CORR-00531-12616/NK29-CORR-00531-13044, Action Item 2015-07-7231: CNSC Type II Compliance Inspection Report: BRPD-AB-2015-008 Implementation of the Bruce Power Reliability Program, Bruce Power Letter, F. Saunders to K. Lafrenière, February 12, 2016.

[198] CAN/CSA-N285.5-08, Periodic Inspection of CANDU Nuclear Power Plant Containment Components, January 2011.

[199] NK21-CORR-00531-10978/NK29-CORR-00531-11366, Approach to Fitness for Service Assessment for Pressure Tubes, Bruce Power Letter, F. Saunders to R. Lojk, December 11, 2013.

[200] NK21-CORR-00531-11159/NK29-CORR-00531-11564, Approach to Fitness for Service Assessment for Pressure Tubes - New Action Item 1407-4775, CNSC Letter, K. Lafrenière to F. Saunders, February 28, 2014.

[201] NK21-CORR-00531-12662/NK29-CORR-00531-13098, Action Item 1407-4775: CNSC Response to Semi-Annual Update #3 on Approach to Fitness-for-Service Assessment for Pressure Tubes, CNSC Letter, K. Lafrenière to F. Saunders, February 29, 2016.

[202] NK21-CORR-00531-12618/NK29-CORR-00531-13046, Action Item 1407-4775: Semi-Annual Update on Approach to Fitness-for-Service Assessment for Pressure Tubes - Update 4, Bruce Power Letter, F. Saunders to K. Lafrenière, February 18, 2016.

[203] NK21-CORR-00531-11273/NK29-CORR-00531-11682, Bruce A and B: Compliance Plan on Long-Term Use of CSA N285.8 – Update, Bruce Power Letter, F. Saunders to K. Lafrenière, June 4, 2014.

[204] NK21-CORR-00531-12172/NK29-CORR-00531-12590, Bruce Units 3 to 8 Updated Deterministic Pressure Tube Fracture Protection Evaluations and Probabilistic Fracture Protection Evaluations, Bruce Power Letter, F. Saunders to K. Lafrenière, July 16, 2015.

[205] NK21-CORR-00531-12107/NK29-CORR-00531-12514, 2015 Annual COG Research and Development Reporting, Bruce Power Letter, F. Saunders to K. Lafrenière, June 12, 2015.

Appendix A – High-Level Assessments Against Relevant Codes and Standards

A.1. CSA-N285.4, Periodic Inspection of CANDU Nuclear Power Plants (NPP)

CSA-N285.4, Periodic Inspection of CANDU Nuclear Power Plant Components is invoked by Condition 6.1, *Fitness for Service*, of the Power Reactor Operating Licence (PROL) [1]. The 2009 version with the 2011 Update [39] is included in the PROL [1]. Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL, and the 2009 version is subject to a transition plan. However, since the latest version of this standard was issued in 2014, this appendix presents a high level code-to-code comparison between the 2014 and 2009 Update No. 2 versions.

The major changes or additions to CSA-N285.4-14 include the following new requirements:

- Clause 12.5, Material surveillance of fuel channel annulus spacers
- Annex H, New informative guidance for preparation of a technical justification for exemption from requirements for steam generator surveillance tube removals

Clause 12.5, 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.

Ageing of annulus spacers is addressed in the Fuel Channel Life Cycle Management Plan, B-PLAN-31100-00001 and the Fuel Channel Condition Assessment, B-REP-31100-00003. Spacer integrity is affected mainly by neutron irradiation, imposed loads, and cyclic loading due to rolling during operation. Spacer movement is a concern because it can lead to Pressure Tube-Calandria Tube (PT-CT) contact and, in the presence of sufficiently high deuterium (D) concentrations, hydride blister formation. Changes in D uptake rate are a concern in this situation because this affects the predictions of the time at which contacting PTs become susceptible to blister formation, and therefore the time at which Spacer Location and Repositioning (SLAR) maintenance is required. After SLAR, monitoring is required to ensure spacers remain in the same location and sag rates remain in the anticipated range to avoid PT-CT contact late in life. The specific requirements (Clause 12.5) 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. This is identified as Issue SF4-2 in Table 10.

Annex H, Guidance for Preparation of a Technical Justification for Exemption from Requirements for Steam Generator Surveillance Tube Removals

Annex H is an informative non-mandatory annex which provides guidance for the preparation of a technical justification for exemption from requirements for steam generator surveillance tube removals.

A.2. CSA-N285.5, Periodic Inspection of CANDU Nuclear Power Plant Containment Components

As discussed in Section 3.2, CSA-N285.5-08, Periodic Inspection of CANDU Nuclear Power Plant Containment Components is invoked under Condition 6.1, *Fitness for Service*, of the PROL [1]. However, the latest version of this standard is N285.5-13. Therefore, this appendix presents a high level code-to-code comparison between the 2013 and 2008 versions.

The major differences between N285.5-08 [198] and N285.5-13 [28] are a new clause 4.6.3 and two new annexes A and B.

Clause 4.6.3 in N285.5-13 states: "In cases when this Standard is being applied to an existing plant or to an existing periodic inspection program written to an earlier edition of CSA-N285.5, the updated program documents shall identify a) the requirements in this Standard that cannot be practically implemented; and b) measures taken to compensate for the requirements that cannot be practically implemented." A footnote clarifies that this Clause is intended to address cases where the inspection program elements specified in Clause 4.6.2 are fundamentally changed in a new edition of CSA-N285.5.

If and when N285.5-13 is applied to Bruce B, compliance with this clause will be required; however, it does not impose any new requirements that would affect ageing management.

Annex A of N285.5-13 provides guidance on periodic inspection, material property monitoring, and test programs for fibreglass reinforced plastics containment components as required by Clause 8.2. This annex is informative and non-mandatory, and compliance is only required if users of this Standard or regulatory authorities adopt it formally as additional requirements.

Annex B of N285.5-13 is an informative guide for periodic inspection and provides the rationale behind the requirements of the standard. It is not a mandatory part of the standard.

It is concluded that the differences between the current version in the PROL and the newer version do not result in any gaps.

A.3. CSA-N285.8, Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors

CSA-N285.8-15 [45] Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in Canada Deuterium Uranium (CANDU) Reactors is the third edition of this standard. It supersedes the previous editions, published in 2010 and 2005. As its name implies CSA-N285.8 is a highly technical standard that specifies mandatory technical requirements and non-mandatory evaluation procedures for fitness-for-service assessments of pressure tubes. Pressure tubes in CANDU nuclear power plants are inspected in accordance with CSA-N285.4. When a detected flaw indication does not satisfy the criteria of acceptance by examination, or when pressure tube to calandria tube contact is detected or predicted, Clause 12 of CSA-

N285.4 permits a fitness-for-service assessment to determine acceptability. Also, Clause 12 of CSA-N285.4 requires evaluation of the results of specified material property surveillance measurements. The evolution of CSA-N285.8 over time represents the results of industry research and development efforts and the increasing sophistication of probabilistic and deterministic assessment methods for pressure tube flaws.

CSA-N285.8-10 [159] provided expanded guidance on probabilistic evaluation of pressure tube degradation mechanisms. A code-to-code comparison of CSA-N285.8-10 [159] with the previous version of this standard, CSA-N285.8-05 [158], revealed the following significant changes:

- Clause 5.2.3.3 provides additional guidance for the characterization of volumetric flaws, specifically with respect to bearing pad fretting flaws;
- Clause 5.4.3.5 addresses explicit process-zone evaluation and added a clause for flaw-tip hydride non-ratcheting conditions (5.4.3.5.3) to the previously included flaw-tip hydride ratcheting conditions (5.4.3.5.2);
- Clause 8 provides technical requirements that shall be satisfied when the results of material surveillance measurements of hydrogen equivalent concentration, fracture toughness, delayed hydride cracking (DHC) growth rate, or threshold isothermal stress intensity factor for DHC initiation do not satisfy the acceptance criteria in Clause 12.4.5 of CSA-N285.4. It proceeds to provide guidance on the use of statistical methods to evaluate the following measurements against the original data set.
    - Fracture toughness (8.3).
    - Delayed hydride cracking growth rate (8.4).
    - Threshold stress intensity factor for delayed hydride cracking (8.5).

The latest version of the standard, CSA-N285.8-15 [45] supplements the information in the 2010 version in the following areas:

- Statistically based fatigue crack initiation evaluation curves for axial flaws. Clauses revised: D.4.2, D.4.3, and D.3.6.
- Closed-form engineering relation for threshold peak stress for DHC initiation. Clauses revised: A.6.3.4.5.1, D.5 and 5.4.3.4.
- Implement statistically based threshold relation for peak stress for crack initiation due to hydrided region overloads. Clauses revised: 3.2, 5.4.3.6, A.6.3.6, and D.5.
- New fracture toughness models for axial through-wall flaws. Clauses revised: D.13.2.
- Material property functional dependencies. Clauses revised: C.4.2.1, C.4.2.2.4 and D.13.3.
- Inconsistency between the caption and the drawing of figures with cross-section view. Clauses revised: Figures 4, A-3 and A-4.
- Implementation Methods 1 and 2 for the probabilistic leak-before-break criterion. Clauses revised: 3.1, 7.3, 7.4 and C.4.3.

With the exception of editorial improvements most of the technical improvements affect the non-mandatory evaluation procedures in the appendices. The technical changes to the mandatory requirements involve:

- The addition of equations to Clause 5.4.3.6 to clarify the requirements for the evaluation of hydrided region overload condition; and,
- The addition of requirements to Clause 7.4.2 and 7.4.3 relating to the justification of specific inputs and assumptions used in the evaluation criteria for application of the leak-before-break criterion.

While important, these mandatory technical changes are explanatory in nature and rather than adding new methods and procedures to the standard, the latest technical changes are designed to clarify and explain methods that were already in the 2010 version.

Bruce Power has been actively striving to become fully compliant with the standard. In December 2013 Bruce Power submitted its approach to fitness for service assessment for pressure tubes to the Canadian Nuclear Safety Commission (CNSC) under cover of NK29-CORR-00531-11366 [199]. The CNSC accepted the proposed approach under several conditions and opened Action Item 1407-4775 (see NK29-CORR-00531-11564 [200]) in which Bruce Power was requested to report semi-annually on progress on the following:

- Revised acceptance criteria for pressure tube failure probability;
- Validation of fracture toughness models; and,
- Application of Probabilistic Leak-Before-Break (PLBB) methodologies, specifically:
    - Treatment of uncertainties;
    - Inter-dependence and cross correlation of parameters;
    - Convergence of Monte Carlo simulations; and
    - Conservatism in postulating the through wall crack in PLBB Method 1.

In their response to the third semi-annual update submitted under this action item (see [201]) the CNSC reduced the frequency of reporting to once a year. Update four, the first annual report, was submitted in February of 2016 under cover of [202].

In 2014 Bruce Power submitted report B-REP-31100-00010, Evaluation Process of Pressure Tube Fitness-for-Service Using CSA-N285.8 [160] to the CNSC as well as a compliance plan [203] for the long term use of CSA-N285.8 for the fitness-for-service assessments pursuant to Licence Condition 6.1 and CSA-N285.4 Clause 12. In support of this compliance plan Bruce Power has been submitting updated deterministic and probabilistic assessments of pressure tube fitness-for-service for Bruce Units 3 to 8. The latest updated assessment was submitted under cover of [204] in July 2015. The compliance plan has recently been updated and the updated version accepted by the CNSC ([160] and [161]).

In addition Bruce Power has been submitting detailed disposition reports for flaws found during inspections based on the application of the methods in CSA-N285.8-10 [159] and have also provided annual reports on research and development (R&D) progress in the area of fuel channel fitness-for-service (see [205]).

Based on the information provided above it is concluded that the 2015 version of CSA-N285.8 does not affect Bruce Power's current degree of compliance with the standard and that Bruce Power is actively working with other partners in industry to advance the science and implementation of the methods advocated in this standard.

Appendix B – Review Against Codes and Standards

This appendix presents the clause-by-clause assessments that are performed for this Safety Factor. The PSR Basis Document provides the following compliance categories and definitions for clause-by-clause assessments:

- Compliant (C) – compliance has been demonstrated with the applicable clause;
- Indirect Compliance (IC) – Compliance has been demonstrated with the intent of the applicable clause;
- Acceptable Deviation (AD) – Compliance with the applicable clause cannot be demonstrated; however, a technical assessment has determined that the deviation is acceptable. For this case a detailed discussion and explanation shall be included in the PSR documentation;
- Gap – system design and/or operational improvements may be necessary;
- Guidance: A potential programmatic, engineering, analytical or effectiveness gap found against non-mandatory guidance;
- Relevant but not Assessed (RNA) – The particular clause provides requirements that are less strenuous than clauses of another standard that has already been assessed. The definition also includes the guidance portion of clauses in which a gap has already been identified against the requirement;
- Not Relevant (NR) – The topic addressed in the specific clause is not relevant to the safety factor under consideration but may well be assessed under a different Safety Factor; and
- Not Applicable (NA) – The text is not a clause that provides requirements or guidance. Also used if the clause does not apply to the specific facility.

B.1. CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants

In support of the review tasks listed in Section 5 relevant clauses of CNSC REGDOC-2.5.2 have been assessed in Table B1. A more detailed assessment is performed in “Safety Factor 1 – Plant Design”.

Table B1: CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants

Article No.	Clause Requirement	Assessment	Compliance Category
5.2	Appropriate design management shall achieve the following objectives: 1. SSCs important to safety meet their respective design requirements. 2. Due account is taken of the human capabilities and limitations of personnel. 3. Safety design information - necessary for safe operation and maintenance of the plant and for any subsequent plant modifications - is preserved. 4. OLCs are provided for incorporation into the plant administrative and operational procedures. 5. The plant design facilitates maintenance and aging management throughout the life of the plant.	Bullet 5 of this clause now includes reference to aging management. Since the Clause 5.2 addresses design management, Bullet 5 is interpreted as requiring that design processes ensure that maintenance and aging management considerations are taken into account during design activities. This implies that equipment should be designed to meet performance requirements throughout its planned life-cycle and also that the configuration of the equipment is such that it enables maintenance and aging management activities. Although aging management comes under the Equipment Reliability program, BP-PROG-11.01, it is linked to design basis management, as per BP-PROG-10.01, Plant Design Basis Management. Implementing procedure BP-PROC-00335, Design Management specifically includes aging managements considerations specifically as a design input in Section 4.3: “Applicable design inputs such as design bases, design criteria and parameters, aging management considerations,	C

Article No.	Clause Requirement	Assessment	Compliance Category
	6. The results of the hazard analysis, deterministic safety analysis and probabilistic safety assessment are taken into account. 7. Due consideration is given to the prevention of accidents and mitigation of their consequences. 8. The generation of radioactive and hazardous waste is limited to minimum practicable levels, in terms of both activity and volume. 9. A change control process is established to track design changes to provide configuration management during manufacturing, construction, commissioning and operation. 10. Physical protection systems and cyber security programs are provided to address design-basis threats.	performance requirements, regulatory requirements, and codes and standards, shall be identified and documented.” Section 4.4 of the procedure specifically identifies aging management considerations and maintenance strategies as design outputs: “An Assigned Design Engineer (ADE) within the responsible design organization is responsible to ensure that applicable requirements of the Design Specifications and codes and standards, as well as any additional aging management considerations, performance requirements, and other design inputs, are correctly translated into specifications, drawings, design reports, analyses, procedures, instructions, and maintenance, testing, and inspection strategies (output documents).“ In addition, implementing procedure BP-PROC-00363, Nuclear Safety Assessment, takes into account the effects of aging.	
5.7	Design documentation shall include information to demonstrate the adequacy of the design and shall be used for procurement, construction, commissioning and safe operation, including maintenance, aging management, modification and eventual decommissioning of the NPP.	The introductory paragraph in this clause is new and includes reference to aging management. The design documentation follows well established processes and procedures as described in Design Documentation, BP-PROC-00335. This procedure	C

Article No.	Clause Requirement	Assessment	Compliance Category
	The design documentation shall include: 1. design description 2. design requirements 3. classification of SSCs 4. description of plant states 5. security system design, including a description of physical security barriers and cyber security programs 6. operational limits and conditions 7. identification and categorization of initiating events 8. acceptance criteria and derived acceptance criteria	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. Under the Equipment Reliability Program, BP-PROG-11.01, life cycle management integrates aging 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. As described in BP-PROC-00400 Life Cycle Management for Critical SSCs, the author of a Life Cycle Management Plan (LCMP) reviews relevant documentation including design requirements and design descriptions when preparing or revising the LCMP. In addition, design changes described in design documentation can trigger a review of LCMPs.	

Article No.	Clause Requirement	Assessment	Compliance Category
	9. deterministic safety analysis 10. probabilistic safety assessment (PSA) 11. hazard analysis Guidance A suite of design documentation should be developed, following the establishment of an overall baseline, listing all key design documents. Design documents should be contained in a logical and manageable framework. For additional guidance on derived acceptance criteria, refer to CNSC regulatory document REGDOC-2.4.1, Deterministic Safety Analysis. Additional information Additional information may be found in: • CNSC, RD/GD-369, Licence Application Guide: Licence to Construct a Nuclear Power Plant,		

Article No.	Clause Requirement	Assessment	Compliance Category
	Ottawa, Canada, 2011. • CNSC, REGDOC-2.4.1, Deterministic Safety Analysis, Ottawa, Canada, 2014.		
7.5	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	Guidance includes aging management to be included as one of the design rules as applicable. The Plant Design Basis Management Program, BP-PROG-10.01, ensures that the plant design meets safety, reliability and regulatory requirements. BP-PROC-00363, Nuclear Safety Assessment, is an implementing procedure under this program which takes into account the effects of aging. The Nuclear Safety Assessment process ensures 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.	C

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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.		
7.8	The design shall include an equipment environmental qualification (EQ) program. Development and implementation of this program shall ensure that the following functions can be carried out:	A new requirement has been added regarding consideration of aging effects due to service life for SSCs important to safety. The Environmental Qualification (EQ) process described in BP-PROC-00261 supports the Design Management procedure BP-PROC-00335 and	C

Article No.	Clause Requirement	Assessment	Compliance Category
	1. the reactor can be safely shut down and kept in a safe shutdown state during and following AOOs and DBAs 2. residual heat can be removed from the reactor after shutdown, and also during and following AOOs and DBAs 3. potential for release of radioactive material from the plant can be limited, and the resulting dose to the public from AOOs and DBAs can be kept within the dose acceptance criteria 4. post-accident conditions can be monitored to indicate whether the above functions are being carried out The environmental conditions to be accounted for shall include those expected during normal operation, and those arising from AOOs and DBAs. Operational data and applicable design assist analysis tools, such as the probabilistic safety assessment, shall be used to determine the envelope of environmental conditions. The equipment qualification program for SSCs	provides assurance that credited essential equipment and components can perform their safety-related functions if exposed to harsh environmental conditions resulting from Design Basis Accidents, in accordance with the plant design and licensing basis and that this capability is preserved over the life of the plant. Aging mechanisms considered in the process include thermal aging, radiation aging and cyclic aging.	

Article No.	Clause Requirement	Assessment	Compliance Category
	important to safety shall include the consideration of aging effects due to service life. Equipment qualification shall also include consideration of any unusual environmental conditions that can reasonably be anticipated, and that could arise during normal operation or AOOs (such as periodic testing of the containment leak rate). Equipment and instrumentation credited to operate during DECs shall be demonstrated, with reasonable confidence, to be capable of performing their intended safety function(s) under the expected environmental conditions. A justifiable extrapolation of equipment and instrumentation behaviour may be used to provide assurance of operability, and is typically based on design specifications, environmental qualification testing, or other considerations. Guidance The designer should provide detailed processes and specifications for an equipment EQ program, for qualifying safety-related equipment associated with systems that are essential to perform the credited safety functions. The EQ program should address		

Article No.	Clause Requirement	Assessment	Compliance Category
	qualification criteria and methods used, and all anticipated environmental conditions upon which the qualification of the equipment (mechanical, electrical, I&C and certain post accident monitoring) is based. The designer should identify the EQ-related standards and codes (e.g., CSA, IEEE and ASME). The latest editions of the applicable standards for use in the equipment qualification are preferred; any deviations should be justified. As a minimum, the basic EQ program elements should be provided as described below. Identification of equipment requiring harsh environmental qualification The design should identify: • systems and equipment required to perform safety functions in a harsh environment, including their safety functions and applicable DBAs • non-safety-related equipment whose failure due to harsh post-accident environment could prevent safety-related equipment from		

Article No.	Clause Requirement	Assessment	Compliance Category
	accomplishing its safety function • accident monitoring equipment Identification of equipment service conditions Service conditions should be identified to determine required qualification methods as they apply to various types of qualification (e.g., harsh environments, mild environments, radiation-only harsh environments). The design should provide for: • a distinction between mild and harsh environments (e.g., specific criteria to define plant environments as either mild or harsh) • a list of bounding harsh DBAs for qualification of equipment • the environmental conditions (e.g., temperature, pressure, radiation, humidity, steam, chemicals, submergence) for each applicable DBA to which equipment is exposed in various plant locations • temperature, pressure and radiation profiles for harsh environment qualification		

Article No.	Clause Requirement	Assessment	Compliance Category
	• typical equipment mission time during DBAs • mild environmental conditions (e.g., temperature, pressure, humidity, radiation) for operational states, including the assumed duration of the AOOs to which equipment is exposed in various plant locations Qualification methods The design should describe methods used to demonstrate the performance of safety-related equipment when subjected to a range of environmental conditions during operational states or DBAs. The methods should determine whether equipment should be qualified for mild or harsh environments. For harsh environment qualification, the design should include the following: • For equipment and components located in a DBA harsh environment, type tests are the preferred method of qualification (particularly for electrical equipment) of qualification; where type tests are not feasible, justification by analysis or operating experience (or a combination of both) may be used. • Equipment should be reviewed in terms of		

Article No.	Clause Requirement	Assessment	Compliance Category
	design, function, materials and environment, to identify significant aging mechanisms caused by operational and environmental conditions occurring during normal operation. Where a significant aging mechanism is identified, that aging should be taken into account in the equipment qualification. • The qualification should systematically address the sequence of age conditioning, including sequential, simultaneous, synergistic effects, and the method for accelerating radiation degradation effects. • Appropriate margins, as given in EQ-related standards, should be applied to the specified environmental conditions. • For certain equipment (e.g., digital I&C equipment, and new advanced analog electronics) additional environmental conditions – such as electromagnetic interference, radio frequency interference, and power surges – should be addressed. For mild environment qualification, equipment may be considered qualified, provided that: • the environmental conditions are specified in a design specification • the manufacturer provides certification that the equipment meets the specification		

Article No.	Clause Requirement	Assessment	Compliance Category
	Equipment and instrumentation credited under design extension conditions A demonstration of equipment and instrumentation operability should include the following: • the accident timeframes for each function • the equipment type and location used to perform necessary functions in each timeframe • the functions credited in the accident timeframes that need to be performed to achieve a safe shutdown state for DECs • the postulated harsh environment of DECs within each timeframe • a reasonable assurance that the equipment will survive to perform its function in the accident timeframes, in the DEC environment Protective barriers The design should address protective barriers, if applicable. When protective barriers are designed to isolate equipment from possible harsh environmental conditions, the barriers themselves should be addressed in a qualification program.		

Article No.	Clause Requirement	Assessment	Compliance Category
	Examples of protective barriers include: • steam-protected rooms and enclosures • steam doors • water-protected rooms (for flooding) Additional information Additional information may be found in: • ASME, QME-1, Qualification of Active Mechanical Equipment Used in Nuclear Power Plants, New York, 2002. • CSA Group, N290.13, Environmental qualification of equipment for CANDU nuclear power plants, Toronto, Canada. • Electric Power Research Institute (ERPI), Technical Report rev. 1, Nuclear Power Plant Equipment Qualification Reference Manual, Palo Alto, California, 2010. • IAEA, Safety Reports Series No. 3, Equipment Qualification in Operational Nuclear Power Plants: Upgrading, Preserving and Reviewing, Vienna, 1998. • International Electrotechnical Commission		

Article No.	Clause Requirement	Assessment	Compliance Category
	(IEC), 60780 ed 2.0, Nuclear Power Plants - Electrical Equipment of the Safety System – Qualification, Geneva, 1998. • IEEE, Standard 323, IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations, Piscataway, New Jersey, 2003. • IEEE, Standard 627, Qualification of Equipment Used in Nuclear Facilities, Piscataway, New Jersey, 2010.		
7.15.1	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	Guidance is provided to consider the impact of aging on the structure and its material during structural design. Structural design considers the impact of aging on structures and materials through the Plant Design Basis Management Program, BP-PROG-10.01 and its implementing procedures. In addition, the Life Cycle Management Plan for Civil Structures, B-PLAN-20000-00001, describes how system performance monitoring, which includes a review of the original design and subsequent modifications, is used to monitor aging degradation for civil structures.	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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.		

Article No.	Clause Requirement	Assessment	Compliance Category
	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)		

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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) Additional information Additional information may be found in:		

Article No.	Clause Requirement	Assessment	Compliance Category
	• American Concrete Institute (ACI), 349-06, Code Requirements for Nuclear Safety-Related Concrete Structures & Commentary, Farmington Hills, Michigan, 2007. • ASME, Boiler and Pressure Vessel Code (BPVC) Section III, Division 2, Section 3, Code for Concrete Containments, New York, 2010. • IAEA, NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants, Vienna, 2004. • U.S. NRC, NUREG/CR-6486, Assessment of Modular Construction for Safety-Related Structures at Advanced Nuclear Power Plants, Washington, D.C., 1997. • U.S. NRC, Regulatory Guide 1.76, Design Basis Tornado and Tornado Missiles for Nuclear Power Plants, Washington, D.C., 2007. • U.S. NRC, Regulatory Guide 1.91, Evaluations of Explosions Postulated to occur on Transportation Routes near Nuclear Power Plants, Washington, D.C., 1978. • U.S. NRC, NUREG-0800, Section 3.8.1, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition - Concrete Containment, Washington, D.C., 2007.		

Article No.	Clause Requirement	Assessment	Compliance Category
7.17	The design shall take due account of the effects of aging and wear on SSCs. For SSCs important to safety, this shall include: 1. an assessment of design margins, taking into account all known aging and wear mechanisms and potential degradation in operational states, including the effects of testing and maintenance processes 2. provisions for monitoring, testing, sampling, and inspecting SSCs so as to assess aging mechanisms, verify predictions, and identify unanticipated behaviours or degradation that may occur during operation, as a result of aging and wear Additional requirements are provided in RD-334, Aging Management for Nuclear Power Plants. Guidance The design should also consider the following: • identification of all SSCs subject to aging management	A new sentence has been added to this clause to refer to additional requirements in RD-334 for Aging Management of Nuclear Power Plants. The Equipment Reliability Program, BP-PROG-11.01, ensures that all systems important to safety (per RD/GD-98) meet their design intent and performance criteria. The scoping and identification of critical SSCs is part of the Equipment Reliability Program implementation. BP-PROC-00778 describes the process for the Responsible System Engineer, with support from Reactor Safety, Corporate & Station Component Engineers and Design Engineering (including Environmental Qualification); to identify SSCs important to maintaining safe, reliable power operation. All aspects of nuclear safety (reactor safety, industrial safety, environmental safety and radiation safety) are addressed. This procedure includes a functional criticality analysis and identifies: - Scoping criteria. - Functions related to safety and reliability. - Critical structures and components that support these functions. - Non-critical components. - Run to failure components.	C

Article No.	Clause Requirement	Assessment	Compliance Category
	• use of advanced materials with greater aging resistant properties • need for materials testing programs to monitor aging degradation • need to incorporate online monitoring, particularly where this technology would provide forewarning of degradation leading to failure of SSCs, and where the consequences of failure could be significant to safety	BP-PROC-00778 uses the Master Equipment List (MEL) as a basis. Components and structures not on the MEL (such as piping, cables, and supports), shall also be reviewed to identify any that are important to maintaining safe, reliable power operation. Data stewardship and governance of the MEL is described in BP-PROC-00584, PASSPORT Equipment Data Management. DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology, determines which plant systems meet the criteria of 'Systems Important to Safety' (SIS). This determination is based on screening criteria which assesses probabilistic risk assessment (PRA) based risk significance, and on non PRA-based system importance for preventing fuel damage and release of radioactivity. The SIS list is used as one of the inputs into the scoping and identification of critical systems. Long Term Planning and Life Cycle Management are specifically discussed in section 4.1.6 of Equipment Reliability Program, BP-PROG-11.01.	
8.1.1	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	Following is guidance related to plant aging excerpted from clause 8.1.1: "The demonstration of thermal margin is expected to be presented in a manner that accounts for all	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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	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." BP-PROC-00363, Nuclear Safety Assessment, takes into account the effects of aging and ensures the safety analysis provides a basis for safe operation. The impact of the condition of the pressure tubes on the thermal margin has been taken into account with new bundle designs such as the modified 37-element (37M) fuel bundle, and the consequences of this have been factored into the safety analyses (NK21-CORR-00531-09574). In addition, analysis of the accidents impacted by ageing are revised to reflect plant conditions applicable to the licence duration. The most recent ageing analyses to 2019 are documented in NK21-CORR-00531-10943/NK29-CORR-00531-11325.	

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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.		

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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.		

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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. Additional information Additional information may be found in:		

Article No.	Clause Requirement	Assessment	Compliance Category
	• ANSI/ANS, 57.5, Light Water Reactor Fuel Assembly Mechanical Design and Evaluation, La Grange Park, Illinois, 1996. • CNSC, G-144, Trip Parameter Acceptance Criteria for the Safety Analysis of CANDU Nuclear Power Plants, Ottawa, Canada, 2006. • U.S. NRC, NUREG-0800, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition - Fuel System Design, Section 4.2, Washington, D.C., 2007.		
8.2	The design shall provide the reactor coolant system (RCS) and its associated components and auxiliary systems with sufficient margin to ensure that the appropriate design limits of the reactor coolant pressure boundary are not exceeded in operational states or DBAs. The design shall ensure that the operation of pressure relief devices will not lead to significant radioactive releases from the plant, even in DBAs. The RCS shall be fitted with isolation devices to limit any loss of radioactive coolant outside containment. The material used in the fabrication of the component parts shall be selected so as to minimize corrosion and activation of the material.	There is a design requirement to take into account all conditions of the boundary material in normal operation (including maintenance and testing), Anticipated Operational Occurrences (AOOs), Design Basis Accidents (DBAs) and Design Extension Conditions, as well as expected end-of-life properties affected by ageing mechanisms, the rate of deterioration, and the initial state of the components. The Plant Design Basis Management Program, BP-PROG-10.01, ensures that the plant design meets safety, reliability and regulatory requirements, including pressure boundary quality assurance requirements as defined in the Pressure Boundary Quality Assurance Program, BP-PROG-00.04.	C

Article No.	Clause Requirement	Assessment	Compliance Category
	Operating conditions in which components of the pressure boundary could exhibit brittle behaviour shall be avoided. The design shall take into account all conditions of the boundary material in normal operation (including maintenance and testing), AOOs, DBAs and DECs, as well as expected end-of-life properties affected by ageing mechanisms, the rate of deterioration, and the initial state of the components. The design of the moving components contained inside the reactor coolant pressure boundary, such as pump impellers and valve parts, shall minimize the likelihood of failure and associated consequential damage to other items of the reactor coolant system. This shall apply to operational states and DBAs, with allowance for deterioration that may occur in service. The design shall provide a system capable of detecting and monitoring leakage from the reactor coolant system. Guidance	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. Engineering analyses performed within the scope of the Equipment Reliability Program and Design Basis Management Program consider ageing mechanisms, the rate of deterioration, and the initial state of the components to assure that SSCs remain within the design and operating envelope over their intended service life. Ageing mechanisms for Structures, Systems, and Components (SSCs) are identified in Life Cycle Management Plans. In particular, ageing mechanisms for Primary Heat Transport (PHT) feeder piping are identified in B-LCM-33126-00001, and ageing mechanisms for fuel channels are identified in B-PLAN-31100-00001.	

Article No.	Clause Requirement	Assessment	Compliance Category
	The design should have adequate provisions with regards to RCS and reactor auxiliary systems. The design should meet design limits for the worst conditions encountered in normal operation, AOOs and DBAs, including pressurized thermal shock and water hammer loads. The RCS and reactor auxiliary systems should meet – or contribute to meeting – the following objectives: • maintain sufficient reactor coolant inventory for core cooling both in and after all postulated initiating events considered in the design basis • remove heat from the core after a failure of the reactor coolant pressure boundary, in order to limit fuel damage • remove heat from the core in appropriate operational states, DBAs and DECs with the reactor coolant pressure boundary intact • transfer heat from other safety systems to the ultimate heat sink The design of each reactor auxiliary system should ensure that automatic action by the system cannot impair a safety function. The design authority should demonstrate the adequacy of the following:		

Article No.	Clause Requirement	Assessment	Compliance Category
	• flow rate and pressure drops across major components • major thermalhydraulic parameters, such as operating pressure and temperature ranges • valve performance (flow, pressure drop, opening and closing times, stability, water-hammer) • pump performance (head, flow, two-phase flow, seal performance) • vibration of components and pipes • control of gas accumulation (in particular, prevention of combustible gas accumulation) • maximum allowable heat-up and cool-down rates • consideration of pressurized thermal shock due to operation (including inadvertent operation) of auxiliary systems • flow stability, including loop-to-loop stability and void-enthalpy oscillations (CANDU) • design of instrumentation taps The following provides a few examples of design expectations of the RCS and reactor auxiliary systems:		

Article No.	Clause Requirement	Assessment	Compliance Category
	Pressurizer For designs that include a pressurizer, the design authority should demonstrate the adequacy of the following: • volume and capability to accommodate load changes, and to accommodate secondary side transients without the need for pressure relief to the containment to the extent practicable • capability to withstand thermal shock, particularly in spray nozzles and connections to the main RCS circuit • control of pressure, such as via heaters, sprays, coolers or steam bleeding Primary pressure relief The design authority should demonstrate the adequacy of the following: • flow rate in single and two phase flow • consideration of corrosion of valve surfaces • provisions for ensuring that relief discharge		

Article No.	Clause Requirement	Assessment	Compliance Category
	does not lead to an unacceptable harsh environment inside containment • relief valve stability Primary reactor coolant pumps For designs that use forced primary flow, the design authority should demonstrate the adequacy of the following: • primary pump performance characteristics, including head and flow characteristics, flow coastdown rate, single and two-phase pump performance • pump operating parameters (e.g., speed, flow, head) • pump net positive suction head needed to avoid cavitation • pump seal design and performance (including seal temperature limitations, if applicable) • vibration monitoring provisions Additional information		

Article No.	Clause Requirement	Assessment	Compliance Category
	Additional information may be found in: • IAEA, NS-G-1.9, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants Safety Guide, Vienna, 2004.		
8.4.1	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 ageing. This	This clause includes a new requirement to take plant aging into account in trip coverage. The effectiveness of trip parameters is addressed through safety analysis performed in accordance with CNSC REGDOC-2.4.1 Deterministic Safety Analysis. The procedure on Nuclear Safety Assessment (NSA) [BP-PROC-00363], defines the elements, functional requirements, implementing procedures and key responsibilities associated with the NSA process. It states that 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. Plant operating limits and conditions are taken into account in the analysis assumptions and inputs of part 3 of the Safety Report. Analysis of accidents impacted by aging are revised to reflect plant conditions	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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 CNSC 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.	applicable to the licence duration. The results of new analysis are consistently used to confirm the adequacy of the Operational Limits and Conditions (OLCs) and if necessary used to derive a more suitable value for use as an operating limit.	

Article No.	Clause Requirement	Assessment	Compliance Category
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 CNSC REGDOC-2.4.1, Deterministic Safety		

Article No.	Clause Requirement	Assessment	Compliance Category
	Analysis. CNSC 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.		
9.2	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	This clause includes new requirements to account for postulated aging effects and demonstrate sufficient design margins. The procedure on Nuclear Safety Assessment (NSA), BP-PROC-00363 ensures 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 safety analyses are based on the as built station. Ageing effects are taken into account, usually by undertaking analyses with expected end-of-life values, confirmed by observing changes in analysis input parameters such as the Reactor Inlet Header (RIH) temperature, PT diametral creep, etc. over	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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	the years. The condition of the pressure tubes has been taken into account with new bundle designs such as the modified 37-element (37M) fuel bundle, and the consequences of this have been factored into the safety analyses. Plant operating limits and conditions are taken into account in the analysis assumptions and inputs of part 3 of the Safety Report. Analysis of the main events impacted by ageing are revised to reflect plant conditions applicable to the licence duration. The results of new analysis are consistently used to confirm the adequacy of the OLCs and if necessary used to derive a more suitable value for use as an operating limit.	

Article No.	Clause Requirement	Assessment	Compliance Category
	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.		

B.2. Incremental Clause-by-Clause Assessment of CSA-N287.1-14, General Requirements for Concrete Containment Structures for Nuclear Power Plants

In support of the review tasks listed in Section 5, a code-to-code comparison has been performed for CSA-N287.1-14 to the version assessed previously for Bruce A (CSA-N287.1-M93) in Table C1. An incremental verification of these new requirements has been performed in Table B2.

Table B2: Incremental Clause-by-Clause Assessment of CSA-N287.1-14, General Requirements for Concrete Containment Structures for Nuclear Power Plants

Article No.	Clause Requirement	Assessment	Compliance Category
4.4.1	Design, fabrication, construction, inspection, examination, and testing shall consider the effects of aging on the containment structure. Note: Aging degradation effects include loss of prestressing force, corrosion, cracking, increased permeability, change in material properties, loss of bond, etc.	Licence Condition 5.1 of the PROL specifies that the licensee shall implement and maintain a design program. The LCH notes that implementing and maintaining a design program confirms that safety-related SSCs and any modifications to them, continue to meet their design bases given new information arising over time and taking changes in the external environment into account. It also confirms that SSCs continue to be able to perform their safety functions under all plant states. Design program is per BP-PROG-10.01, Plant Design Basis Management. This program ensures the design basis provides a basis for safe operation, and includes consideration of ageing management. Structures are fabricated and installed in accordance with established procedures, e.g., nuclear construction requirements manual. Although aging management is not specifically	C

Article No.	Clause Requirement	Assessment	Compliance Category
		addressed, care and attention to good fabrication and construction practices are inherent to minimize the impact of construction on the component. Where there are particular requirements, these will be contained in the procedures used. An example would be foreign material exclusion (FME); eliminating FME provides assurances that construction practices will not be life limiting. Inspection and testing of containment structures are part of the PROL. B-PLAN-20000-00001 is the Life Cycle Management Plan for Civil Structures. This plan identifies cracking and corrosion as the most reported degradation mechanisms for civil structures. The plan also addresses the following forms of degradation caused by transport mechanisms within the pores and cracks and the presence of water: - Chemical attack from sulphates, acids and bases, alkali aggregate and carbonation. - Physical attack from leaching, elevated temperature, the crystallization of chlorides and other salts, abrasion/erosion, irradiation, fatigue/vibration and settlement, excessive thermal stress at attachments and in embedded cooling circuits. - shrinkage, wet/dry cycling, freeze/thaw cycling and	

Article No.	Clause Requirement	Assessment	Compliance Category
		acid rain. - loss of pre-stress in post tensioned concrete members.	
4.4.2	An appropriate margin shall be provided in the design, taking into account relevant aging mechanisms and the potential for age-related degradation in normal operation and accident scenarios.	Containment leakage testing and inspections are the primary methods for monitoring degradation of civil structures. Acceptance criteria and safety margins are described in B-PLAN-20000-00001, Life Cycle Management Plan for Civil Structures. The operational target for the Bruce NGS A and B main containment structure is 1.0% contained mass/hr at the design pressure of the structure. However, the Operating Policies and Principles value is 2.0%/hr at the design pressure of the structure. As described in B-PLAN-20000-00001, the CANDU industry has developed methods to address leaks in containment concrete. These methods are described in DPT-MP-00005, Negative Pressure Containment Structure Concrete Repair. DPT-MP-00005 also incorporates relevant material from Chapter 5 of the International Atomic Energy Agency (IAEA) report TECDOC 1025 on the assessment and repair of ageing effects in concrete containment buildings.	C
4.4.3	The design should enable the assessment of aging.	Bruce Power's design basis management program BP-PROG-10.01, Plant Design Basis Management ensures the design basis provides a basis for safe operation. The design enables the assessment of aging through the testing of containment boundary	C

Article No.	Clause Requirement	Assessment	Compliance Category
		leakage.	
4.4.4	An aging management program shall be established that enables assessment of the ability of the containment structure to satisfy the functional requirements specified in the design documentation and safety report for the life of the plant. Notes: 1) The aging management program can be a mix of surveillance, testing, or other methods that provide assurance of sustained performance. 2) In-service examination and testing form part of the aging management program.	Bruce Power has established an overall Aging Management program framework which is governed by BP-PROG-11.01, Equipment Reliability. Relevant procedures are BP-PROC-00778, Scoping and Identification of Critical SSCs, BP-PROC-00779, Continuing Equipment Reliability Improvement, BP-PROC-00781, Performance Monitoring, and BP-PROC-00783, Long Term Planning & Life Cycle Management. For civil structures, ageing degradation is monitored using the following methods: - visual inspections - leak rate tests - pre-stressing force determinations - system performance monitoring Ageing monitoring for containment structures specifically will be based on the Periodic Inspection Program (PIP) results obtained in accordance with N285.5 and N285.7, leakage rate test results, Plant Health reports, Station Condition Records (SCRs), Operating Experience (OPEX), etc. It should be noted that inspection and testing of containment structures are part of the PROL, and specified limits are included in the Operating Policies and	C

Article No.	Clause Requirement	Assessment	Compliance Category
		Principles.	
4.4.5	The results from the aging management program implementation shall be used to assess the rate of degradation and to adjust the aging and maintenance programs accordingly.	As part of the Life Cycle Management Plan for Civil Structures, B-PLAN-20000-00001, Condition Assessments are required for critical civil structure components (as listed in Table 4 of B-PLAN-20000-00001). These Condition Assessments describe component ageing degradation, ageing monitoring and an ageing mitigation outline. The Condition Assessments also address Feedback Mechanisms (i.e. Linking the inspection findings back into the LCMP/CA program).	C

B.3. CSA-N291-15, Requirements for Safety-Related Structures for Nuclear Power Plants

In support of the review tasks listed in Section 5 relevant clauses of CSA-N291 have been assessed in Table B3. A high-level assessment of the complete standard is performed in "Safety Factor 1– Plant Design".

Table B3: CSA-N291-15, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants

Article No.	Clause Requirement	Assessment	Compliance Category
7.3.2	**In-service examination program**		
7.3.2.1	The engineer shall establish an in-service examination program to provide assurance of structures sustained performance. The in-service examination program shall cover general requirements for examination of safety-related structures and their components. The examination program shall include requirements for additional examination of critical components identified by the designer in accordance with Clause 7.1.2. The in-service examination program will support the aging management plan as detailed in Clause 9.	Bruce Power is developing an in-service examination program document for safety related structures. As an interim transition measure for compliance with N291-08, Bruce Power has developed CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures, NK29-PIP-20000-00001, to document existing inspections.	C
7.3.2.2	The examination program shall include the following: a) scope of examination; b) general examination requirements; c) identified degradation mechanisms;	The scope of CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures, NK29-PIP-20000-00001, includes the following: 1) Responsibilities within the Bruce Power organization for preparation of this program, performance of the in-service inspections, testing	C

Article No.	Clause Requirement	Assessment	Compliance Category
	d) locations of components to be examined; e) methods of examination and testing; f) frequency and amount (i.e., statistical distribution) of examination and testing; g) acceptance criteria; and h) reporting and documentation requirements. Note: Special consideration should be given to factors such as accessibility, effectiveness, and accuracy of examination methods.	and reporting of results 2) General requirements for personnel qualifications, basis of comparisons and repairs/replacements and modifications 3) Frequency of examination 4) Identification of areas and/or components to be inspected 5) Means of investigation, procedures/tests, acceptance criteria and remedial actions 6) Reporting and documentation requirements. The program description also addresses signs of degradation in several places	
7.3.2.3	The extent of examination and basis of comparison shall be established with consideration to the following: a) importance of the structure, element, or component; and b) elastic deformation and distortion of the structure, with particular emphasis on those points where maximum structural movement or stress is expected. Notes: 1) Points of measurement and inspection should be similar to those used in previous examinations, where possible, to facilitate comparison of results and trending	CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures, NK29-PIP-20000-00001, defines inspection and reporting guidelines for the safety related structures identified in Section 5.0, including: Reactor Auxiliary Bay and Secondary Control Area (0,3,4) Accumulator Building and ECI Service Bridge ECI Storage Tank, Service Area, and Pipe Tunnel Powerhouse Turbine Tables CCW Piping, Piping Supports, Discharge Duct, and Outfall Structure Pumphouses Cooling Water Intake Tunnel and Intake Structure Recirculation Duct & Control Structure	C

Article No.	Clause Requirement	Assessment	Compliance Category
	2) Basis of comparison should be in accordance with Clause 5.2 of CSA-N287.7 as applicable.	Primary Irradiated Fuel Bay Service Building Ancillary Service Building Secondary Irradiated Fuel Bay Construction Retube Building and Secondary Control Area (1,2) EFADS Building Old Water Treatment Plant, QPS Room, and Access Tunnel Standby Generator Buildings and Oil Pumphouse Miscellaneous Steel Structures The areas and/or components to be inspected are identified in Section 5.0 of NK29-PIP-20000-00001. General inspection criteria include visual inspection for: - water ingress - bent, twisted, deformed or missing structural members - bolted connections not tight - cracks in steel members or welds - outside building siding intact, check for missing fasteners - signs of corrosion - concrete degradation	

Article No.	Clause Requirement	Assessment	Compliance Category
		- leaking or broken window and door seals - condition of doors, windows and framing - condition of irradiated fuel storage bay liners	
7.3.3.1	Accessible surfaces of safety-related structures shall be examined at least once every 6 years, subsequent to the first in-service use of the plant, in accordance with the in-service examination program for the safety-related structures. For components not normally accessible, the examinations shall be at a frequency agreed upon by the owner/operator and the AHJ. Note: For large surface areas, an in-service examination plan may consist of sample areas that are representative of limiting conditions within the structure. The areas should be selected based on the factors that can affect integrity (e.g., radiation, temperature, and high stress).	Section 4.5 of NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures states that: "Accessible surfaces of safety-related structures shall be examined at least once every 6 years". The inspection schedule is presented in Appendix A of NK29-PIP-20000-00001.	C
7.3.3.2	The frequency of examinations shall be increased for components or parts that have exhibited significant deterioration and that might warrant frequent future repair or replacement.	Section 4.0 of NK29-PIP-20000-00001, CSA-N291 In-Service Inspection Program for Bruce NGS B Safety Related Structures states that: "All safety related structures shall be visually examined. The examination shall be of sufficient frequency and physical extent to define any significant changes or degradation".	C

Article No.	Clause Requirement	Assessment	Compliance Category
7.3.4	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.	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
9	**Aging Management**		
9.1	An aging management plan shall be developed, implemented, and maintained by the owner/operator to provide for the timely detection and mitigation of aging effects to ensure integrity and functional capability of the structure throughout all stages of its life cycle including design, construction, commissioning, operation, and decommissioning. The aging management plan shall be submitted to the authority having jurisdiction (AHJ) for acceptance. Notes: 1) In Canada, CNSC RD-2.6.3 provides the regulatory requirements for managing the aging of structures, systems, and components of a nuclear power plant (NPP). 2) CSA-N287.8 can be used as guidance for development and implementation of aging management plan for safety-related concrete structures. 3) The aging management plan can include surveillance, testing, or other methods that provide	A number of Civil structures at Bruce B have been identified in the Safety Related System List, BP-PROC-00169, as structures whose degradation or failure could have serious safety or economic consequences. The Equipment Reliability Program B-PROG-11.01 requires that all SSCs identified as such be part of the Ageing Management program, requiring the preparation of an LCMP in accordance with BP-PROC-00400. Bruce Power has consequently developed and implemented a LCMP for Civil Structures, B-PLAN-20000-00001. It describes industry best practice in understanding ageing degradation of civil structures, and best practice for detection and mitigation. Acceptance criteria and required safety margins are discussed as these provide a basis for remaining life assessment of the structure. The plan was submitted to the CNSC in response to Action Item 090708 via NK29-CORR-00531-08849 in July 2010. After the CNSC's comments were addressed the plan was accepted by the CNSC via	C

Article No.	Clause Requirement	Assessment	Compliance Category
	assurance of a structure's sustained performance. 4) In-service inspection, examination, and testing activities form part of the aging management plan.	NK29-CORR-00531-09877 Feb of 2012	
9.2	The results from the aging management plan implementation shall be used to assess the rate of degradation and to adjust the aging and in-service examination and testing programs accordingly.	The Life-Cycle Management Plan for Civil Structures, B-PLAN-20000-00001, requires that Condition Assessments be developed for structures that are shown to be critical to safety and generation. Results from the In-Service Inspection program documented in NK29-PIP-20000-00001 potentially results in updates to the Life-Cycle Management Plan via Performance and Condition Monitoring (BP-PROC-00382). The latter requires that systems and performance monitoring plans be created to track and assess system or component performance. In addition, system/component health reports are prepared to document performance monitoring results and trending and provide a system health status.	C
9.3	Design, fabrication, construction, operation, and decommissioning shall consider the effects of aging on the safety-related structures.	Aging management comes under the Equipment Reliability program, BP-PROG-11.01, and is linked to design basis management, as per BP-PROG-10.01, Plant Design Basis Management. Specifically, implementing procedure BP-PROC-00363, Nuclear Safety Assessment, takes into account the effects of aging.	C
9.4	A safety margin shall be provided in the design, taking into account relevant aging mechanisms and the potential for age-related degradation in normal	The Plant Design Basis Management Program, BP-PROG-10.01, ensures that the plant design meets safety, reliability and regulatory requirements. BP-	C

Article No.	Clause Requirement	Assessment	Compliance Category
	operation and accident scenarios.	PROC-00363, Nuclear Safety Assessment, is an implementing procedure under this program which takes into account the effects of aging. The Nuclear Safety Assessment process ensures 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.	
9.5	The design should be provided to facilitate inspection, examination, testing, surveillance, maintenance, repair, and replacement activities, and to keep potential radiation exposures during these activities as low as reasonably achievable.	For an existing plant like Bruce B, the focus is to keep potential radiation exposures as low as reasonably achievable. The As Low As Reasonably Achievable (ALARA) Program as documented in BP-RPP-00044 is in place to ensure this objective.	C

Appendix C – Code-to-Code Comparison for Updated Codes and Standards

C.1. Comparison of CSA-N287.1-14, General Requirements for Concrete Containment Structures for Nuclear Power Plants to CSA-N287.1-M93 (R2014), General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants

In support of the review tasks listed in Section 5, a code-to-code comparison has been performed for CSA-N287.1-14 to the version assessed previously for Bruce A (CSA-N287.1-M93). CSA-N287.1-14 clauses without equivalent clauses in CSA-N287.1-M93 have been identified in Table C1. An incremental clause-by-clause assessment of these new requirements has been performed in Appendix B.2, within Table B2.

Table C1: Code-to-Code Comparison of CSA-N287.1-14 to CSA-N287.1-M93 (R2014)

Clause	Clause Text	Associated Clause(s)	Assessment	Evaluation
4.4			This clause is a section heading and has not been assessed.	New Requirement
4.4.1	Design, fabrication, construction, inspection, examination, and testing shall consider the effects of aging on the containment structure. Note: Aging degradation effects include loss of prestressing force, corrosion, cracking, increased permeability, change in material properties, loss of bond, etc.		This clause presents a new requirement.	New Requirement

Clause	Clause Text	Associated Clause(s)	Assessment	Evaluation
4.4.2	An appropriate margin shall be provided in the design, taking into account relevant aging mechanisms and the potential for age-related degradation in normal operation and accident scenarios.		This clause presents a new requirement.	New Requirement
4.4.3	The design should enable the assessment of aging.		This clause presents a new requirement.	New Requirement
4.4.4	An aging management program shall be established that enables assessment of the ability of the containment structure to satisfy the functional requirements specified in the design documentation and safety report for the life of the plant. Notes: 1) The aging management program can be a mix of surveillance, testing, or other methods that provide assurance of sustained performance. 2) In-service examination and testing form part of the aging management program.		This clause presents a new requirement.	New Requirement
4.4.5	The results from the aging management program implementation shall be used to assess the rate of degradation and to adjust the aging and maintenance programs accordingly.		This clause presents a new requirement.	New Requirement