FORM-14159 R000*

Periodic Safety Review - Final Document Review Traveler

Bruce Power

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Bruce Power Document Title:
Safety Factor 4 – Ageing

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

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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	27AUG2015

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

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Reviewed By:

Name	Title	Department	Signature	Date
Ken Chuong	Sr. Technical Specialist	Nuclear Safety Analysis and Support		July 27, 2015
Patrick Ta	Sr. Technical Specialist	Nuclear Safety Analysis and Support	Patrick Ta	July 27, 2015
Vlad Costiuc	Sr. Design Engineer	Nuclear Safety Analysis and Support		Jul 27, 2015
David Cho	Sr. Technical Specialist	Nuclear Safety Analysis and Support		July 27, 2015
Andrew Glover	Sr. Technical Engineer	Nuclear Safety Analysis and Support		27 JUL 2015

Recommended for Use By:

Name	Title	Department	Signature	Date
Phil Hunt	Division Manager	Reactor Safety Engineering	P. Hunt	19 Aug 2015

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Reviewed By:

Name	Title	Department	Signature	Date
Susan Brissette	Dept. Manager	Management Systems	S Brissette	31 JUL 15

Recommended for Use By:

Name	Title	Department	Signature	Date
Susan Brissette	Dept. Manager	Management Systems	S Brissette	31 JULY 15

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Reviewed By:

Name	Title	Department	Signature	Date
Rob McLennan	Division Manager	Nuclear Programs	Rd McLennan	18Aug 2015

Recommended for Use By:

Name	Title	Department	Signature	Date
Rob McLennan	Division Manager	Nuclear Programs	Rd McLennan	18Aug 2015

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Reviewed By:

Name	Title	Department	Signature	Date
Bogdan Melnikov	Sr. Technical Engineer	Major Components Engineering	[signature]	29JUL15
Richard Xamin	Engineering Trainee	Major Components Engineering	[signature]	29JUL2015
Stephen Miller	Department Manager	Equipment Reliability Integration	[signature]	19AUG2015

Recommended for Use By:

Name	Title	Department	Signature	Date
Kevin Pickles	Division Manager	Station Engineering	[signature]	20AUG2015

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Reviewed By:

Name	Title	Department	Signature	Date
Steven Ostrowski	Department. Manager	Component Design & Design Programs		21 AUG 2015
Jim Coady	Department. Manager	I&C and Electrical		21 AUG 2015

Recommended for Use by:

Name	Title	Department	Signature	Date
Gord Kozak	Division Manager	Engineering Support		24 AUG 2015

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Reviewed By:

Name	**Title**	**Department**	**Signature**	**Date**

Recommended for Use By:

Name	**Title**	**Department**	**Signature**	**Date**

Division of Kinectrics Inc.

Title: Safety Factor 4 - Ageing

File: K-421231-00014-R00

NK21-SFR-09701-00004

A Report Submitted to Bruce Power

June 30, 2015

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<tr><td>Author:
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G. Archinoff
T. Kapaklili</td><td>Approver:</td><td>Date:
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L. Watt</td><td>Approver:</td><td>Date:
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<tr><td>Author:
J. Ling
C. Oberholzer</td><td>Verifier:
G. Buckley</td><td>Reviewer:
G. Archinoff
L. Watt</td><td>Approver:</td><td>Date:
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J. Ling
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C. Stallman</td><td>Reviewer:
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Acronyms and Abbreviations

AFCC	As Found Condition Coordinator
ALPO	Asset Life Projections and Options
AMOT	Asset Management Options Template
AMP	Ageing Management Program
Bruce 1&2	Bruce Units 1 and 2
BP	Bruce Power
CANDU	Canada Deuterium Uranium
CAPE	Component and Program Engineering
CCP	Critical Channel Power
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
DEM	Duty Engineering Manager
DHC	Delayed Hydride Cracking
EA	Environmental Assessment
ELCE	Equipment Life Cycle Engineering
EFPH	Equivalent Full Power Hours
EQ	Environmental Qualification
ER	Equipment Reliability
EWMS	Engineering Work Management System
FAC	Flow Accelerated Corrosion
FASA	Focused Area Self-Assessment
FCAMP	Fuel Channel Ageing Management Plan
FCCA	Fuel Channel Condition Assessment
FCLCMP	Fuel Channel Life Cycle Management Plan
FFS	Fitness for Service

FMEA	Failure Modes and Effects Analysis
FRP	Fibreglass Reinforced Plastics
GE	General Electric
HTS	Heat Transport System
IAEA	International Atomic Energy Agency
INPO	Institute of Nuclear Power Operations
ISR	Integrated Safety Review
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
MCR	Major Component Replacement
MEL	Master Equipment List
NGS A	Nuclear Generating Station A
NOP	Neutron Overpower
NPP	Nuclear Power Plant
NSA	Nuclear Safety Assessment
NSCA	Nuclear Safety and Control Act
OBSE	Obsolescence (action plan type)
OFI	Opportunities for Improvement
O&M	Operations and Maintenance
OPEX	Operating Experience
OP&P	Operating Policies and Principles
PdM	Predictive Maintenance
PHT	Primary Heat Transport
PI	Performance Indicator
PIP	Periodic Inspection Program
PM	Preventive Maintenance
PMEL	Performance Monitoring Equipment List

PMIDRQ	Predefine Identification Requirement
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSR	Periodic Safety Review
PT-CT	Pressure Tube-Calandria Tube
R&D	Research & Development
RCM	Risk Control Measures
RIHT	Reactor Inlet Header Temperature
RS	Reactor Safety
RSE	Responsible System Engineer
SBLOCA	Small Break Loss-of-Coolant Accident
SBR	Safety Basis Report
SCR	Station Condition Record
SFR	Safety Factor Report
SG	Steam Generator
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
WO	Work Order

1. Objective and Description

Bruce Power (BP), as an essential part of its operating strategy, is planning to continue operation of Units 3 and 4 as part of its contribution to the Long Term Energy Plan (LTEP) (http://www.energy.gov.on.ca/en/ltep/). Bruce Power has developed plant life integration management plans in support of operation to 247,000 Equivalent Full Power Hours (EFPH). A more intensive Asset Management program is under development, which includes a Major Component Replacement (MCR) approach to replace 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 maintenance outage will be scheduled concurrently.

To support the definition and timing of practicable opportunities for enhancing the safety of Units 3 and 4, and the ongoing operation of Units 1 and 2, which have already been refurbished, Bruce Power is conducting a station-wide review of safety for Units 0A and 1-4, to be termed an Integrated Safety Review (ISR) [1]. This ISR supersedes the Bruce A portion of the interim Periodic Safety Review (PSR) that was conducted for the ongoing operation of the Bruce A and B units until 2019 [2]. This ISR is conducted in accordance with the Bruce A ISR Basis Document [1], which states that the ISR will meet or exceed the international guidelines given in International Atomic Energy Agency (IAEA) Guide SSG-25, Periodic Safety Review for Nuclear Power Plants [3]. The ISR envelops the guidelines in Canadian Nuclear Safety Commission (CNSC) Regulatory Document RD-360 [4], Life Extension for Nuclear Power Plants, with the exception of those related to the Environmental Assessment (EA), which has already been completed for Bruce A [5].[1]

1.1. Objective

The overall objective of the Bruce A ISR is to conduct a review of Bruce A against modern codes and standards and international safety expectations and provide input to a practicable set of improvements to be conducted during the Major Component Replacement in Units 3 and 4, and during asset management activities to support ongoing operation of all four units, including U0A, that will enhance safety to support long term operation. The look-ahead period will be longer than that in the interim PSR performed for Units 1-8 [2]. 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. Nuclear Safety is a primary consideration for Bruce Power and the management system must support the enhancement and improvement of safety culture and the achievement of high levels of safety, as well as reliable and economic performance.

[1] RD-360 [4] was superseded by CNSC REGDOC-2.3.3 [6] in April 2015. REGDOC-2.3.3 was in draft at the time that the ISR Basis Document [1] was prepared. The draft version of REGDOC-2.3.3 stated that it was consistent with SSG-25, and the assessments in the Safety Factor Reports were performed on that basis. The issued version of CNSC REGDOC-2.3.3 also states that it is consistent with SSG-25, and therefore it is considered that the ISR envelops the guidelines in CNSC REGDOC-2.3.3.

The specific objective of the review of this Safety Factor is to determine whether ageing[2] 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 over the full operational life of the plant.

1.2. Description

The review is conducted in accordance with the Bruce A ISR Basis Document [1], which states that the review tasks are as follows:

1. The following programmatic and technical aspects of the ageing management program are addressed:
    a. The timely detection and mitigation of ageing mechanisms and/or ageing effects;
    b. The comprehensiveness of the program, i.e., does it address all SSCs important to safety?
    c. The effectiveness of operating and maintenance policies and/or procedures for managing the ageing of replaceable components;
    d. Evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs important to safety;
    e. Management of the effects of ageing on those parts of the nuclear power plant that will be required for safety when the nuclear reactor has ceased operation, for example the spent fuel storage facilities;
    f. Performance indicators;
    g. Record keeping.
2. The review addresses the following technical aspects:
    a. Ageing management methodology;
    b. The operating organization's understanding of dominant ageing mechanisms and phenomena, including knowledge of actual safety margins;
    c. Availability of data for assessing ageing degradation, including baseline data and operating and maintenance histories;
    d. Acceptance criteria and required safety margins for SSCs important to safety;
    e. Operating guidelines aimed at controlling and/or moderating the rate of ageing degradation;
    f. Methods for monitoring ageing and for mitigation of ageing effects;
    g. Awareness of the physical condition of SSCs important to safety and any features that could limit service life;
    h. Understanding and control of ageing of all materials (including consumables, such as lubricants) and SSCs that could impair their safety functions; and
    i. Obsolescence of technology used in the nuclear power plant.

[2] In this Safety Factor Report, "ageing" and "aging" are used interchangeably. Bruce Power documents generally use "aging" while the IAEA's SSG-25 [3] uses "ageing".

2. Methodology of Review

As discussed in the Bruce A ISR Basis Document [1], the methodology for an ISR should include making use of safety reviews that have already been performed for other reasons. Accordingly, the Bruce A ISR 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 (circa 2006) [8] [9];
- Proposed refurbishments of Bruce Units 3 and 4 (circa 2008) [10] [11] [12]; and
- Safety Basis Report (SBR) and Periodic Safety Review (PSR) for Bruce Units 1 to 8 (2013) [2].

These reviews covered many, if not all, of the same Safety Factors that are reviewed in the current ISR. A full chronology of Bruce Power safety reviews is provided in Appendix F of [13].

The Bruce A ISR 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 ISR Basis 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 ISR 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 3.
3. **Determine the type and scope of assessment to be performed:** This step involves confirming or modifying the assessment type for each of the codes and standards and guidance documents identified for consideration. The ISR 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 ISR Basis Document.

4. **Perform gap assessment against codes and standards:** This step comprises the actual assessment of the Bruce Power programs and the Bruce A plant against the identified codes and standards. In general, this involves determining from available design or programmatic documentation whether the plant's design or programs 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 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 gap 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.
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 provide a cross section, intended to 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 A 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. The list also includes any new codes or standards that came into effect after the completion of the 2013 PSR, as well as those that supersede codes or standards previously assessed. Regulatory codes and standards issued after the code effective date of August 31, 2014 were not considered in the review.

3.1. Acts and Regulations

The *Nuclear Safety and Control Act* (NSCA) [14] establishes the Canadian Nuclear Safety Commission and its authority to regulate nuclear activities in Canada. The NSCA has been amended on July 3, 2013 to provide the CNSC with the authority to establish an administrative monetary penalty system. The Administrative Monetary Penalties Regulations were introduced in 2013, and set out the list of violations that are subject to administrative monetary penalties, as well as the method and criteria for penalties administration. However, these changes do not impact this Safety Factor. Furthermore, following the Fukushima nuclear events of March 2011, the Fukushima Omnibus Amendment Project was undertaken and completed in 2012, and resulted in amendments to regulatory documents to reflect lessons learned from these events. Bruce Power has a process to ensure compliance with the NSCA [14] 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 Bruce Power Reactor Operating Licence (PROL) [15] and Licence Conditions Handbook (LCH) [16] noted in Table C-1 of the ISR Basis Document [1] are identified in Table 1.[3] The edition dates referenced in the third column of the table are the modern versions used for comparison.

The PROL [15] contains several conditions related to ageing management:

- Licence condition 4.1 requires the licensee to establish, document and implement a maintenance program in accordance with S-210 “Maintenance Programs for Nuclear Power Plants”
- Licence condition 4.2 requires that the licensee plan and execute outage maintenance in accordance with a program applicable to all work affecting the continued safe operation of the facility.
- Licence condition 4.3 (i) requires that the licensee implement and maintain a periodic inspection and testing program in accordance with the following Canadian Standards Association (CSA) standards:

[3] PROL 18.00/2020 [17] and LCH-BNGS-R000 [18] came into effect on June 1, 2015. However, PROL 15.00/2015 [15] and LCH-BNGSA-R8 [16] are the versions referred to in this ISR, as these were in force when the assessments in the Safety Factor Reports were performed.

- N285.4: Periodic Inspection of CANDU Nuclear Power Plant Components
- N285.5: Periodic Inspection of CANDU Nuclear Power Plants Containment Components; and
- N287.7: In-service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants.

- Licence condition 4.3 (ii) requires that the licensee implement and maintain an in-service inspection program for the safety significant balance of plant systems, structures and components;
- Licence condition 4.4 requires that the licensee implement and maintain a reliability program in accordance with CNSC regulatory document S-98 entitled: Reliability Programs for Nuclear Power Plants;

In addition, Licence Condition 1.7 requires that the licensee notify and report in accordance with CNSC regulatory document CNSC REGDOC-3.1.1, which includes reporting on metrics related to pressure boundary degradation, plant reliability and preventive maintenance.

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

Document Number	Document Title	Modern Version Used for ISR Comparison	Type of Review
CNSC RD-360	Life Extension of Nuclear Power Plants	CNSC RD-360 (2008) [4]	NR
CNSC S-98	Reliability Programs for Nuclear Power Plants	CNSC RD/GD-98 [19]	NR
CNSC S-99	Reporting Requirements for Operating Nuclear Power Plants	CNSC REGDOC-3.1.1 [20]	NR
CNSC S-210	Maintenance Programs for Nuclear Power Plants	CNSC RD/GD-210 [21]	NR
CSA-N285.4-05	Periodic Inspection of CANDU Nuclear Power Plant Components	CSA-N285.4-14 [22]	CTC (HL)
CSA N285.5-08	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	CSA N285.5-13 [23]	CTC (HL)
CSA N286-05	Management System Requirements for Nuclear Power Plants	CSA N286-12 [24]	NR

Document Number	Document Title	Modern Version Used for ISR 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) [25]	NR
CSA N290.13-05	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	CSA N290.13-05 (R2010) [26]	NR
Assessment type: Clause-by-Clause (CBC); Code-to-Code (CTC); High Level (HL); No Assessment Required (NR); Confirm Validity of Previous Assessments (CV)			

CNSC RD-360: This ISR is being conducted as part of ongoing operation for Units 1 and 2 and to support Major Component Replacement of Units 3 and 4, so it also envelops the guidelines in RD-360, Life Extension for Nuclear Power Plants, issued February 2008. Therefore, RD-360 [4] *de facto* continues to provide guidance on how this review should be conducted. However, RD-360 [4] was superseded by CNSC REGDOC-2.3.3 [6] in April 2015, which was in draft at the time that the ISR Basis Document [1] was prepared. The draft version of CNSC REGDOC-2.3.3 stated that it was consistent with SSG-25, and the assessments in the Safety Factor Reports were performed on that basis. The issued version of CNSC REGDOC-2.3.3 also states that it is consistent with SSG-25, and therefore it is considered that the ISR envelops the guidelines in CNSC REGDOC-2.3.3.

CNSC RD/GD-98: Table C-1 of the ISR Basis Document [1] calls for a confirmation of validity of the previous reviews of Regulatory document RD/GD-98 [19], Reliability Programs for Nuclear Power Plants, which sets out the requirements and guidance of the CNSC for the development and implementation of a reliability program for nuclear power plants in Canada. RD/GD-98 [19] captures the existing requirements previously found in the eponymous S-98 (Revision 1) [27] and also replaces the latter document. A review against S-98 was completed for the Bruce 1 and 2 ISR and submitted to the CNSC and the program was established and implemented as required by licence condition 4.4 of the PROL [15]. The ISR Basis Document [1] identified this review as “Confirm Validity of Previous Assessment”. However, RD/GD-98 does not add to the requirements of S-98 [13] and continues to be a licence condition. 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 REGDOC-3.1.1: CNSC REGDOC-3.1.1 [20] was not identified as relevant for Safety Factor 4 in Table C-1 of the ISR Basis Document [1]. CNSC REGDOC-3.1.1 [20], Reporting Requirements for Nuclear Power Plants, which replaced S-99 [28] in May 2014, is listed as Condition 1.7 in the PROL [15] and sets reporting requirements for nuclear power plants including metrics related to pressure boundary degradation, plant reliability and preventive maintenance. Bruce Power switched over to CNSC REGDOC-3.1.1 at the beginning of 2015[4], as committed in a letter submitted to the CNSC [29]. 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: Regulatory document RD/GD-210 [21], Maintenance Programs for Nuclear Power Plants, sets out the requirements of the CNSC with regard to maintenance programs for nuclear power plants. It specifies that a maintenance program consists of policies, processes and procedures that provide direction for maintaining SSCs of the plant. RD/GD-210 [21] replaces regulatory standard S-210 [30] (published in 2007). It reaffirms the existing requirements found in S-210 [30], and 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 [31]. 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 [30] has been included in the licence as condition 4.1. A code-to-code comparison of RD/GD-210 versus S-210 with respect to ageing was performed in 2013 as part of the interim PSR and it was determined that RD/GD-210 [21] does not add to the requirements of S-210 [30]. Bruce Power is fully compliant with RD/GD-210, as noted in Reference [29]. Since RD/GD-210 will be listed in the PROL, line-by-line compliance with this regulatory document is verified on an ongoing basis to ensure compliance with the PROL. Therefore assessment of RD/GD-210 is not included in this Safety Factor.

CSA N285.4-14: CSA N285.4, Periodic Inspection of CANDU Nuclear Power Plant Components is listed as condition 4.3(i) (a) in the PROL [14]. A high level review of the 2005 version of this standard was performed as part of the Bruce 3&4 ISR. A newer version of this standard was issued in 2009 [32] with an Update in 2011; the CNSC has indicated that the 2009 version with the 2011 Update will be included in the next PROL [33]. Since Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL, the 2009 version will be subject to a transition plan and compliance was not assessed as part of this ISR. However, the latest version of this standard is N285.4-14. Therefore, a high level code-to-code comparison between the 2014 and 2009 versions was conducted and the results presented in Appendix A (A.1).

CSA N285.5-13: CSA N285.5-08, Periodic Inspection of CANDU Nuclear Power Plant Containment Components is listed as condition 4.3(i) (b) in the PROL [15], and thus Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL. The latest version of this standard is N285.5-13, which supersedes that of N285.5-08 previously assessed in the interim PSR. The ISR Basis Document [1] identified this review as “Confirm Validity of Previous Assessment”. However, since a newer version of the standard was issued in 2013, a high level code-to-code comparison between the 2013 and 2008 versions was conducted and the results presented in Appendix A (A.2).

CSA N286-12: Table C-1 of the ISR Basis Document [1] calls for a code-to-code review against CSA standard CSA N286-05. 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

[4] Reporting is performed under S-99 up to the end of 2014, and under CNSC REGDOC-3.1.1 for periods thereafter.

Management System" and have acknowledged that "the new requirements in CSA N286-12 are already addressed in Bruce Power's program and procedure documentation" [35].

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 next licensing period [36]. 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 [37]. Bruce Power further stated CSA N286-12 does not establish any significant or immediate new safety requirements that would merit a more accelerated implementation. 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.

CSA N287.7-08: CSA N287.7-08 [25], In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, is listed as condition 4.3(i)(c) in the PROL [15], and thus Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL. There is not a newer version of this standard. The ISR Basis Document [1] identified this review as "Confirm Validity of Previous Assessment". However, since N287.7-08 is listed in the PROL and has not been revised, and Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL, an assessment against N287.7-08 [25] was not performed as part of this Safety Factor.

CSA N290.13-05: CSA N290.13-05 [26], Environmental Qualification of Equipment for CANDU Nuclear Power Plants, is listed in condition 4.5 of the PROL, and therefore Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL. There is not a newer version of this standard. The ISR Basis Document [1] identified this review as "Confirm Validity of Previous Assessment". However, since N290.13-05 has not been revised and is listed in the PROL, Bruce Power verifies line-by-line compliance with this standard on an ongoing basis to ensure compliance with the PROL. Therefore an assessment against N290.13 was not performed as part of this Safety Factor.

3.3. Regulatory Documents

In addition to the Regulatory Documents identified in the Bruce Power PROL [15] and LCH [16] (Table 1), other Regulatory Documents identified in Table C-1 of the ISR Basis Document [1] 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	[38]	CBC
CNSC REGDOC-2.6.3	Fitness for Service: Aging Management	[39]	NR
Assessment type: Clause-by-Clause (CBC); Code-to-Code (CTC); High Level (HL); No Assessment Required (NR); Confirm Validity of Previous Assessments (CV)			

CNSC REGDOC-2.5.2: Table C-1 of the ISR Basis Document [1] does not identify CNSC REGDOC-2.5.2 as relevant to Safety Factor 4; however, upon further consideration review of specific clauses in support of the identified review tasks was completed as documented in Appendix B.

CNSC REGDOC-2.6.3: Table C-1 of the ISR Basis Document [1] calls for a Code-to-Code Assessment of differences between CNSC RD-334 [40] and CNSC REGDOC-2.6.3 [39]. However, Bruce Power completed a gap assessment of Bruce Power governance against CNSC REGDOC-2.6.3, and submitted a transition plan for CNSC REGDOC-2.6.3 implementation [41]. The gap assessment confirmed that the existing governance largely aligns with the requirements of CNSC REGDOC-2.6.3, 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. Therefore, no further assessment of CNSC REGDOC-2.6.3 is necessary in the review of this Safety Factor.

3.4. CSA Standards

Additional CSA standards which are considered for application to review tasks of this Safety Factor are included in Table 3. These include CSA standards that are identified in Table C-1 of the ISR Basis Document [1].

Table 3: CSA Standards

Document Number	Document Title	Reference	Type of Review
CSA N285.8-10	Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	[42]	CV

Document Number	Document Title	Reference	Type of Review
CSA N291-08 (R2013)	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	[44]	CBC
CSA N287.1-14	General Requirements for Concrete Containment Structures for Nuclear Power Plants	[45]	CTC
Assessment type: Clause-by-Clause (CBC); Code-to-Code (CTC); High Level (HL); No Assessment Required (NR); Confirm Validity of Previous Assessments (CV)			

CSA N285.8-10: CSA N285.8-10 [42] Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors is the second edition of this standard, which was first published in 2005 as N285.8-05. The requirements of N285.8 address the specific fitness-for-service evaluation requirements of N285.4, Clause 12. The 2010 version of this standard provides updated requirements for the assessment of pressure tube flaws and therefore a code-to-code comparison between the 2010 and 2005 versions was conducted and the results assessed as part of the interim PSR. The previous code-to-code comparison documented in Section 5.5.1 of Safety Factor 4 for the interim PSR [2] was reviewed and confirmed to be still valid, i.e., no gaps were identified with respect to ageing. As described in the interim PSR [2], the acceptance criteria for Clause 8 of CSA-N285.8-10 [42] continues to develop as newer results from COG Research & Development (R&D) activities become available. Bruce Power is moving towards compliance with CSA-N285.8-10 [42] as outlined in report B-REP-31100-00010, “Evaluation Process of Pressure Tube Fitness-for-Service Using CSA N285.8” [43], which was updated in May 2014.

CSA N291-08: CSA N291-08 [44], Requirements for Safety Related Structures for CANDU Nuclear Power Plants, provides material, design, construction, fabrication, inspection and examination requirements for CANDU safety-related structures. Aspects of this standard related to ageing are assessed in Appendix B (B.3) of this Safety Factor, whereas a comprehensive review of CSA N291-08 is addressed in “Safety Factor 1: Plant Design”.

CSA N287.1-14: CSA N287.1-14 [45], 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.

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	[3]	NR
Assessment type: Clause-by-Clause (CBC); Code-to-Code (CTC); High Level (HL); No Assessment Required (NR); Confirm Validity of Previous Assessments (CV)			

IAEA SSG-25: IAEA SSG-25 [3] addresses the periodic safety review of nuclear power plants and is the governing document for the review of the ISR, as identified in the Bruce A ISR Basis Document [1]. It defines the review tasks that should be considered for this Safety Factor. 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 A Station Programs and Processes

Ageing management for Bruce A & B is governed by a cross-functional collection of governance documents that is mostly centered in the Equipment Reliability (ER) program. This section provides an overview of aging management at Bruce Power, with emphasis on the Equipment Reliability program and the procedures within that program that govern the key elements of ageing management as defined by CNSC REGDOC-2.6.3 [39]. The complete ER program is defined in BP-PROG-11.01 [46]. Bruce Power's Aging Management Roadmap, which follows the PLAN-DO-CHECK-ACT approach in CNSC REGDOC-2.6.3 [39], is provided in BP-PROC-00783, "Long Term Planning & Life Cycle Management" [47], and reproduced in this Safety Factor Report as Figure 1. The Bruce Power programs and procedures relevant to plant ageing are identified in Table 5.[5]

[5] Table 5 lists the key governance documents used to support the assessments of the review tasks for this Safety Factor Report. There is a continual process to update the governance documents; document versions may differ amongst individual Safety Factor Reports depending on the actual assessment review date. A full set of current sub-tier documents is provided within each current PROG document.

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

First Tier Documents	Second Tier Documents	Third Tier Documents	Fourth Tier Documents
BP-MSM-1: Management System Manual [48]	BP-PROG-11.01: Equipment Reliability [46]	BP-PROC-00778: Scoping and Identification of Critical SSCs [49]	BP-PROC-00584: PASSPORT Equipment Data Management [54]
			BP-PROC-00666: Component Categorization [55]
			BP-PROC-00533: Obsolescence Management [56]
			DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology, [57]
		BP-PROC-00779: Continuing Equipment Reliability Improvement [50]	BP-PROC-00532: Critical and Strategic Spares [58]
			BP-PROC-00534: Technical Basis Assessment [59]
			BP-PROC-00383: Performance and Condition Assessment [60]
		BP-PROC-00780: Preventive Maintenance Implementation [51]	BP-PROC-00501: Integrated Preventive Maintenance Program [61]

First Tier Documents	Second Tier Documents	Third Tier Documents	Fourth Tier Documents
			BP-PROC-00457: Development and Approval of Predefined [62]
			BP-PROC-00603: Preventive Maintenance “Just-in-Time” [JIT] Review Process [63]
		BP-PROC-00781: Performance Monitoring [52]	BP-PROC-00284: Predictive Maintenance [64]
			DPT-PE-00008: System/Component Performance Monitoring Plan [65]
			DPT-PE-00010: System Health Reporting [66]
			DPT-PE-00011: Component Health Reporting [67]
		BP-PROC-00782: Equipment Reliability Problem Identification and Resolution [53]	BP-PROC-00496: Trouble Shooting Plant Equipment [68]
			DIV-ENG-00004: Engineering Evaluations [69]
		BP-PROC-00783: Long Term Planning and Life Cycle Management [47]	BP-PROC-00400: Life Cycle Management for Critical SSCs [70]

In addition to these procedures related to Equipment Reliability, Bruce Power has also issued the following Programs and Procedures relevant to this Safety Factor:

- DPT-NSAS-00016, Integrated Aging Management for Safety Assessment, [71].
- BP-PROG-12.02, Chemistry Management, [72].
- BP-PROC-00169, Safety-Related System List, [73].
- BP-PROC-00060, Station Condition Record Process, [74].
- BP-PROC-00019, Action Tracking, [75].
- BP-PROG-11.04, Plant Maintenance, [76].
- DPT-CHM-00003, Control of Chemistry, [77].
- DPT-CHM-00007, Performance Monitoring, [78].
- DPT-CHM-00008, Outage Chemistry Program, [79].

In particular, DPT-NSAS-00016 [71] 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 [72], provides governance for control of chemistry (DPT-CHM-00003 [77]), performance monitoring with respect to chemistry control (DPT-CHM-00007 [78]) and the outage chemistry program (DPT-CHM-00008 [79]). BP-PROG-11.04 [76] governs plant maintenance, including the performance of hands-on-maintenance of plant Structures, Systems and Components (SSC) in accordance with approved maintenance strategies, schedules, procedures and practices.

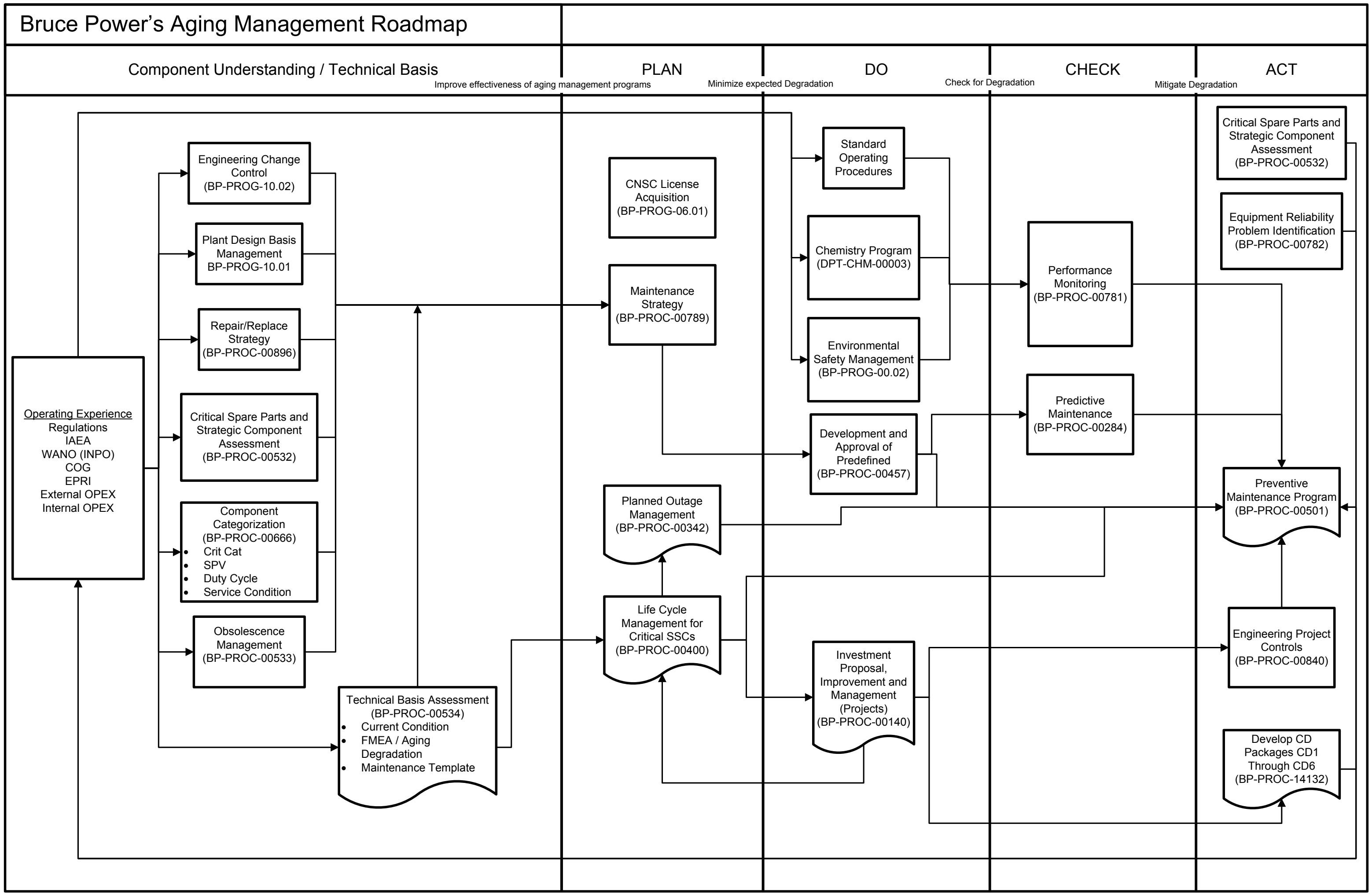

Figure 1: Bruce Power Ageing Management Roadmap

The Equipment Reliability (ER) program provides an overall description of the processes that govern equipment reliability at Bruce Power and establishes a framework to monitor and maintain SSCs in a manner such that nuclear safety, reliability, availability, cost and performance are optimized while ensuring regulatory compliance. The objectives of the ER program [46] are to ensure:

- The process is efficient, incorporates human factor considerations, and ensures effective performance during all phases of plant operations;
- A uniform process is used among all plants in the organization;
- Applicable in-house and industry lessons learned are incorporated into the process to improve adequacy and efficiency; and
- Changes to the process are timely, responsive to user feedback and implemented at all affected plants.

The Equipment Reliability program as defined in BP-PROG-11.01 and its implementing set of procedures is based upon the Institute of Nuclear Power Operations (INPO) Equipment Reliability Advanced Process Description (AP-913). The six implementing procedures of BP-PROG-11.01 are wholly aligned with the 6 sub-processes defined in this industry standard. AP-913 is in turn a sub-process of the larger standard nuclear performance model. These six implementing procedures are:

- BP-PROC-00778, Scoping and Identification of Critical SSCs [49]
- BP-PROC-00779, Continuing Equipment Reliability Improvement [50]
- BP-PROC-00780, Preventive Maintenance (PM) Implementation [51]
- BP-PROC-00781, Performance Monitoring [52]
- BP-PROC-00782, Equipment Reliability (ER) Problem Identification and Resolution [53]
- BP-PROC-00783, Long Term Planning and Life Cycle Management [47]

BP-PROC-00778, Scoping and Identification of Critical SSCs [49], describes the process for identifying SSCs that are important to maintaining safe, reliable operation, and BP-PROC-00779 [50] provides a continuous improvement review process to optimize the technical basis based on station equipment operating experience. BP-PROC-00780, Preventive Maintenance Implementation [51], BP-PROC-00781, Performance Monitoring [52], and BP-PROC-00782, ER Problem Identification and Resolution [53], are interfacing procedures that lead the continuous improvement process. BP-PROC-00783, Long Term Planning and Life Cycle Management [47], enables the development of Life Cycle Management Plans (LCMPs) and the identification and management of obsolescence issues.

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. Scoping and Identification of Critical SSCs

BP-PROC-00778, Scoping and Identification of Critical SSCs [49], describes the process for identifying SSCs that are important to maintaining safe, reliable operation. This procedure identifies scoping criteria, functions of SSCs related to safety and reliability, critical SSCs that support these functions, non-critical components and run-to-failure components.

4.1.1. Identification of Important Functions

Defining the SSC functions that are important to providing safe, reliable power operation is the responsibility of the Responsible System Engineer (RSE) and involves review and evaluation of the following:

- SSCs identified in the Safety Related Systems List (BP-PROC-00169 [73])
- Components identified as Single Point Vulnerability (SPV) (Component Categorization, BP-PROC-00666 [55])
- Systems identified as “important to safety” as defined by the station Probabilistic Risk Assessment (PRA) (DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology [57]), and consistent with the requirements of RD/GD-98 [19].
- Functions identified in the Safety Report, System Design Manuals and station safety analysis
- Regulatory requirements
- Environmental Qualification (EQ) Safety Related Component List

Based on a review of the above information, each system’s important functions are captured in its System/Component Performance Monitoring Plan (SPMP/CPMP) (see DPT-PE-00008 [65]).

4.1.2. Identification of Critical, Non-Critical and Run-to-Failure Components

Component categorization is an input to Continuing Equipment Reliability Improvement (BP-PROC-00779 [50]) and Performance Monitoring (BP-PROC-00781 [52]). The categorization of components is essential to the successful application of maintenance strategies, the selection of performance monitoring criteria and most other aspects of equipment reliability.

Component criticality category (Crit-Cat) is determined based on the functional failure effect of the component and the value of preventive maintenance. A critical component is one whose function is essential to system operation and/or operability (Crit-Cat 1 & 2). Specifically:

- Crit-Cat 1 components significantly contribute to Reactor Safety, Radiological Safety, Environmental Safety, or Employee Safety.
- Crit-Cat 2 components are associated with, but do not significantly contribute to, Reactor Safety, Radiological Safety, Environmental Safety, or Employee Safety.

- Crit-Cat 3 includes non-critical components for which it is more cost-effective to perform preventive maintenance activities.
- Crit-Cat 4 includes run-to-failure components for which it is more cost-effective to perform corrective maintenance activities.

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 [50], and will be subject to monitoring requirements established in BP-PROC-00781 [52]. Crit-Cat 4 components are deemed "run-to-failure" and will not receive any preventive maintenance.

BP-PROC-00666 [55] describes the process of categorizing the components and 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 [50] and BP-PROC-00532 [58].

Service Condition and Duty Cycle are required to support maintenance template development and component level Preventive Maintenance (PM) strategy application (as per BP-PROC-00779 [50] and BP-PROC-00780 [51]. S-98 equipment importance as a designation of risk importance is input into many processes including procurement.

Once the categorization is complete, the designations are documented in PASSPORT in accordance with BP-PROC-00584, PASSPORT Equipment Data Management [54]. RSEs continually monitor these data and updates are completed on an on-going basis.

4.1.3. Identification of Equipment Obsolescence Vulnerabilities

BP-PROC-00533, Obsolescence Management [56] 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. The Obsolescence Management Process also provides provisions for Obsolescence issues as they occur during normal work activities.

The Obsolescence Process Coordinator is responsible for maintaining an awareness of all known obsolescence issues for both Bruce A and Bruce B. These issues are tracked on the Site Obsolescence List.

4.2. Continuing Equipment Reliability Improvement

BP-PROC-00779, Continuing Equipment Reliability Improvement [50], describes the process for development and optimization of the preventive maintenance technical basis and requisite tasks to support a documented PM program for SSCs identified in BP-PROC-00778 [49] to be a part of the ER program.

BP-PROC-00779 [50] also provides a continuous improvement review process to optimize the technical basis based on station equipment operating experience.

This process provides inputs 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.

The review provided by BP-PROC-00779 [50] 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 review takes into account industry experience, new predictive maintenance technologies, changes in degradation at a rate different from that expected, and feedback from maintenance personnel.

BP-PROC-00534, Technical Basis Assessment [59], 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) and results from the Condition Assessments 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 [51] and also captured in BP-PROC-00783, Long Term Planning and Life Cycle Management [47].

In support of Continuing ER Improvement, BP-PROC-00532, Critical and Strategic Spares [58] provides the process for identifying critical parts for SSCs and determining which of those critical parts are to be classified as critical and/or strategic spares.

4.3. Preventive Maintenance Implementation

BP-PROC-00780, Preventive Maintenance Implementation [51], 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-000780 [51] presents the process for capturing information from maintenance personnel on the as-found condition and providing feedback to the Responsible System Engineer/Station Component Engineer.

Once the PM tasks and frequencies are established per BP-PROC-00779 [50] in PASSPORT, the Maintenance PM Assessor will generate a Predefine Identification Requirement (PMIDRQ) from the information provided. BP-PROC-00457 [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 prior to work week execution as per BP-PROC-00603, Preventive Maintenance “Just-in-Time (JIT)” Review Process [63].

BP-PROC-00501, Integrated PM Program [61], provides the methodology to effectively specify PM activities, achieve ER goals and continuously improve the Bruce Power site PM programs.

4.4. Performance Monitoring

BP-PROC-00781, Performance Monitoring [52], provides the basis and expectations for the 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 & Identification of Critical SSCs [49].

BP-PROC-00781 [52] 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.
- Monitoring by Responsible System Engineers/Station Component Engineers walkdowns.

BP-PROC-00781 [52] provides lists of systems and their relative placement in the hierarchy of importance in the definition of the scope or the performance and condition monitoring program.

The performance criteria and monitoring paramaters are obtained from the SPMPs/CPMPs (DPT-PE-00008 [65]) or TBAs prepared in accordance with BP-PROC-00779 [50]. Performance monitoring results are recorded in System Health Reports (SHR) or Component Health Reports (CHR) as per the intervals established in the System Health Reporting procedure, (DPT-PE-00010 [66]) or Component Health Reporting procedure (DPT-PE-00011 [67]).

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

There are many other governing documents that contain elements of system, structure and component performance and condition monitoring. They are tabulated in BP-PROC-00781 [52]. The significant ones are:

- Major Component Outage Scope Management, BP PROC-00017 [80]
- Fuel and Fuel Channel Program, BP-PROC-00893 [81]
- In-Service Testing and Inspection to Satisfy CSA-N287.7-08 Requirements, BP-PROC-00361 [82]
- Performance Requirements for Contamination Exhaust Control Filters, DPT-PE-00005 [83]

Performance Monitoring is suported by BP-PROC-00284, Predictive Maintenance (PdM) [64], which establishes the requirements to implement, maintain and continuously improve the PdM Program by integrating various equipment condition monitoring technologies. Predictive Maintenance (PdM) or condition based maintenance evaluates the condition of equipment by performing periodic or continuous (on-line) equipment monitoring. The ultimate goal of PdM is to perform maintenance “just in time”, before the equipment fails in service.

4.5. Equipment Reliability Problem Identification and Resolution

BP-PROC-00782, Equipment Reliability Problem Identification and Resolution [53], describes the problem resolution process, including the interface with the Station Condition Record (SCR) Process (BP-PROC-00060 [74]) and the Action Tracking Process (BP-PROC-00019 [75]). 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. Required Corrective Maintenance is executed according to the procedures under BP-PROG-11.04, Plant Maintenance program [76].

BP-PROC-00782 [53] provides feedback to developing and implementing long-term system or component health improvement plans. Periodic assessments are made of system, component and program health and vulnerabilities in Health Reports.

The process interfaces with the Plant Health Committee for prioritization of key equipment problems based on safety, operational impact and station availability.

This process describes how equipment reliability improvement results from a low tolerance for equipment problems and a common station focus to completely resolve key equipment problems. Key procedures that support this objective include BP-PROC-00496, Trouble Shooting Plant Equipment [68] and DIV-ENG-00004, Engineering Evaluations [69].

4.6. Long Term Planning and Life Cycle Management

BP-PROC-00783, Long Term Planning and Life Cycle Management [47], 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 Continuous Equipment Reliability Improvement process (BP-PROC 00779 [50]).

The asset management process facilitates business decisions about capital and Operations & Maintenance (O&M) investments, long term planning and asset replacement, and maintenance plans and priorities. The asset management planning process [84] involves a formal presentation and approval of the selected asset management scope.

Asset Life Projections and Options (ALPO) documents are developed to summarize the existing condition of components, mitigation options to optimize the life of the component and the relevant supporting programs to provide an End of Life projection. ALPO documents are developed using BP-PROC-00899, Asset Life Projections and Options [85].

The options from the ALPO document are one of the key inputs for asset management scope selection and approval. Upon approval of the initial or subsequent revisions, contents of the asset management plan will be forwarded to Engineering for incorporation in the Life Cycle Management Plans (LCMP), which are developed using BP-PROC-00400, Life Cycle Management for Critical SSCs [70].

The Corporate Component Engineer works with the Asset Management Group to complete a standard template of information called the Asset Management Options Template (AMOT) regarding the ALPO options for the component. The Component Engineer, along with the Asset Management Group presents the options to the Asset Management Options Selection Committee, which reviews the options presented and either selects an option or combination of options and approves the placement (timing) on the plan, or requests further information prior to option selection.

The asset management model is shown in Figure 2.

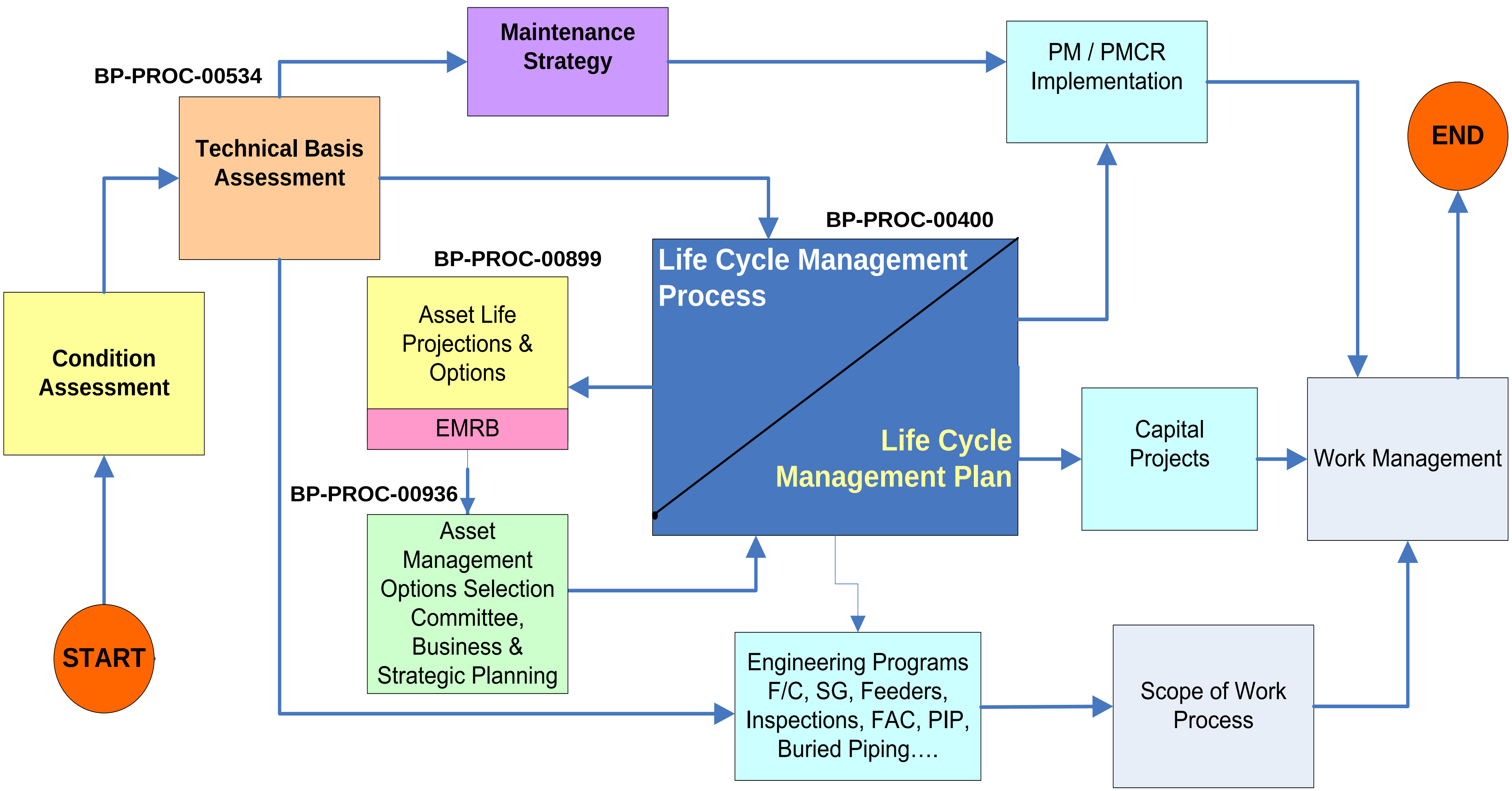

Figure 2: Bruce Power Asset Management Model

The asset management process drives the following processes:

- Strategic and long-range planning
- Generation planning
- Project evaluation and ranking
- Budgeting
- Plant/fleet valuation
- Ageing Management

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 ageing degradation and an estimate of the remaining service life. External and Bruce Power experience is considered in identifying ageing issues. The SSC long-term recommended mitigation options are produced in the Life Cycle Mangement 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 [47]) and Oboslescence Management process (BP-PROC-00533 [56]).

The Life Cycle Management for Critical SSCs (BP-PROC-00400 [70]) provides the basis and expectations for 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 [19]), 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.

5. Results of the Review Tasks

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 depend on an effective PM Program and a Performance Monitoring Program to continuously confirm effectiveness. These programs are supported at Bruce Power by procedures governing Pipe Wall Thinning – Flow Accelerated Corrosion (FAC) (BP-PROC-00923 [86]), Periodic Inspection (BP-PROC-00334 [87]), and Buried Piping Inspection Program (BP-PROC-00825 [88]).

BP-PROC-00923 [86] 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 [87] describes how the requirements for the Periodic Inspection Program of plant structures, systems and components 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 are identified in the Periodic Inspection Plans for Bruce A Units 1 to 4 [89], [90], [91], [92]:

(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 – Fuelling 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 exempt from the basic requirements of the N285.4 Standard based on failure size. However, they do constitute part of a vital system and represent components or materials used beyond conditions of proven experience. As such, they are addressed by supplementary 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):

- The Fuel Channel Life Cycle Management Plan (FCLCMP), B-PLAN-31100-00001 [93],
- Fuel Channel Periodic Inspection Program, B-PIP-31100-00001 [94]

Feeder Pipes (CSA N285.4 Clause 13.0):

- PHT Feeder Piping Life Cycle Management Plan B-LCM-33126-00001 [95]

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

- The Steam Generator and Preheater Life Cycle Management Plan, B-PLAN-33110-00001 [96]

Containment boundary components subject to periodic inspection under CSA N285.5 are identified in NK21-PIP-03642-00001, “Bruce NGS A N285.5 Periodic Inspection Plan for Unit 0A and Units 1 to 4 Containment Components” [97].

BP-PROC-00334 also 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 [88] 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 [51], 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-000780 [51] presents the process for capturing information from maintenance personnel on the as-found condition and providing feedback to the Responsible System Engineer/Station Component Engineer.

Once the PM tasks and frequencies are established per BP-PROC-00779 [50] in PASSPORT, the Maintenance PM Assessor will generate a PMIDRQ from the information provided. BP-PROC-00457 [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 prior to work week execution as per BP-PROC-00603, Preventive Maintenance “Just-in-Time (JIT)” Review Process [63].

BP-PROC-00501, Integrated PM Program [61], 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 [52], provides the basis and expectations for the Equipment Performance Monitoring Process.

Performance Monitoring is supported by BP-PROC-00284, Predictive Maintenance (PdM) [64] 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 [64] invokes *inter alia* the following implementing procedures:

- BP-PROC-00323, Predictive Maintenance Lubrication Analysis [98];
- BP-PROC-00762, Predictive Maintenance Ultrasound Inspection Program [99];
- BP-PROC-00768, Predictive Maintenance Infrared Thermography Program [100];
- SEC-RE-00009, Predictive Maintenance Vibration Monitoring [101]; and
- SEC-RE-00016, Predictive Maintenance Motor Testing Program [102].

Review Task Conclusion

Bruce Power's Preventive Maintenance Program, supported by its various inspection programs and Performance Monitoring Program 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 Program and supporting programs 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 SCCs 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 [49]. 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 [73])
- Components identified as Single Point Vulnerability (SPV) (BP-PROC-00666 [55])
- Systems identified as "important to safety" as defined by the station PRA (DPT-RS-00012, Systems Important to Safety (SIS) Decision Methodology [57])

- Functions identified in the Safety Report, System Design Manuals and station safety analysis
- Regulatory requirements
- 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 [65] (see also BP-PROC-00781 [52]).

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 for components associated with each important function. BP-PROC-00666 [55] 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). Specifically:

- Crit-Cat 1 components significantly contribute to Reactor Safety, Radiological Safety, Environmental Safety, or Employee Safety.
- Crit-Cat 2 components are associated with, but do not significantly contribute to, Reactor Safety, Radiological Safety, Environmental Safety, or Employee Safety.
- Crit-Cat 3 includes non-critical components for which it is more cost-effective to perform preventive maintenance activities.
- Crit-Cat 4 includes run-to-failure components for which it is more cost-effective to perform corrective maintenance activities.

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 [50], and will be subject to monitoring requirements established in BP-PROC-00781 [52]. Crit-Cat 4 components are deemed “run-to-failure” and will not receive any preventive maintenance.

BP-PROC-00666 [55] 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 [50] and BP-PROC-00532 [58].

Service Condition and Duty Cycle are required to support maintenance template development and component level PM strategy application (as per BP-PROC-00779 [50] and BP-PROC-00780 [51]). 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, [103]. 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 [49] 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 failure while others are subject to a performance measurement 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 [104] defines the rules for the management of operations, maintenance and modification work performed during power operation. BP-PROC-00329 [105] 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.

A review of maintenance backlogs was conducted and submitted to the CNSC under cover of NK21-CORR-00531-10769 [106] 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 and BP-PROG-11.02 [104]. Bruce Power is implementing a comprehensive action plan to reduce the backlogs for Elective Maintenance and Preventive Maintenance deferrals. CNSC staff recently conducted a Type II compliance inspection from April to September 2014 on Bruce

Power's management of the Bruce A Unit 3 planned maintenance outage [107]. This inspection confirmed that the Bruce Power outage maintenance backlog reduction met requirements, but found that improvements are needed to drive down the backlogs. In the Quarterly Field Inspection Report for Q2 2014 (July 1, 2014 to September 30, 2014), CNSC staff also 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 [108]. Regular updates will be provided to the CNSC on progress being made, and these updates will continue until the backlogs are reduced to a sustainable level that meets industry standards. This is tracked under Action Item 1307-4113 [109].

The review provided by BP-PROC-00779 [50] 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 [50] 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.

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 [110].

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. However, further improvement is still needed, and is being tracked under Action Item 1307-4113. 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 [52] identifies the following data sources that can be used to assist in performance monitoring activities:

- PM results (as per BP-PROC-00284 [64], 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 Responsible System Engineer/Station Component Engineer as per DPT-PE-00009 [111].
- Operator rounds as per GRP-OPS-00047 [112].
- Inspection results from PASSPORT and/or Resident Inspection.
- Issues/actions raised by Duty Engineering Manager (DEM).
- As Found Condition reports from the As Found Condition Coordinator (AFCC) (“As Found Condition Codes” captured via PM Completion Module in PASSPORT).
- Shift/Outage Logs.
- 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.

BP-PROC-00781 [52] 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 by Responsible System Engineers/Station Component Engineers walkdowns.

The results of monitoring and trending activities are to be captured within the System/Component/Program Health Reports (SHR/CHR) by the associated Responsible System Engineer/Station Component Engineer/Program Engineer as per the intervals established in the System Health Reporting procedure, (DPT-PE-00010 [66]) or Component Health Reporting procedure (DPT-PE-00011 [67]). 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 Responsible System Engineer/Station Component Engineer/Program Engineer 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 [53], Equipment Reliability Problem Identification and Resolution).

The Responsible System Engineer/Station Component Engineer/Program Engineer can determine if there is degraded performance by comparing their monitoring/trending results within their SHR/CHR, against the SPMP/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" captured by the As Found Condition Coordinator [AFCC] are prescribed in BP-PROC-00780 [51], Preventive Maintenance Implementation); or
- Conditional/dynamic data monitoring indicates a degrading trend (from Responsible System Engineer/Station Component Engineer/Program Engineer analysis).

If the results from monitoring/trending activities have identified a degraded SSC condition, the Responsible System Engineer/Station Component Engineer determines if an SCR is required in accordance with the guidance provided in BP-PROC-00782 [53].

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

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

A CNSC report on the application of the CNSC Risk-Informed Decision Making (RIDM) process to Category 3 CANDU Safety Issues (CSIs) identified 16 Category 3[6] issues in 2009 [113],

[6] 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.

including CI1[7] "Fuel Channel Integrity and Effect on Core Internals", PF19 "Impact of Ageing on Safe Plant Operation" and GL3 "Ageing of Equipment and Structures" related to ageing. The CNSC report also identified Risk Control Measures (RCMs) for each CANDU Safety Issue (CSI).

The RCMs for CI1 are:

- Document and implement an integrated Fuel Channel Ageing Management Plan (FCAMP).
- 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 preaccident pressure tubes characteristics.

The RCMs for PF19 are:

- 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.
- Complete condition assessment in the context of plant life extension projects.

Bruce Power addressed the Risk Control Measures associated with these CSIs and requested reclassification of these issues from Category 3 to Category 2 in December 2012 [114]. In April 2013, CNSC staff reclassified PF19 from Category 3 to Category 2 [115]. In October 2013, CI1 was reclassified to Category 2 [116], and in April 2014, GL3 was reclassified to Category 2 [117].

Review Task Conclusion

The System/Component/Program health evaluation and reporting provisions of BP-PROC-00781 [52] 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.

7 Ageing of the fuel channels impact on safe and reliable operation of the plant.

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 [52], 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 [49].

Bruce A 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 [52]. Components and programs scoped into the performance monitoring program are identified in Appendix C to BP-PROC-00781 [52].

The lists of systems and components provided in Appendices B and C of BP-PROC-00781 [52] 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 [3] 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 [52] 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, 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). The RSE then denotes 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 Continuing Equipment Reliability process (BP-PROC-00779 [50]) 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 Station Component Engineer/Program Engineer establishes component performance criteria and monitoring parameters for their responsible component group, and documents, within the PMEL, all the monitoring parameters from the PMs generated from the Continuing Equipment Reliability process (BP-PROC-00779) for the critical and non-critical components identified in PASSPORT, Meridium or the Plant IQ software suite. This provides the monitoring baseline for the Station Component Engineer/Program Engineer.

Then Responsible System Engineer/ Station Component Engineer/Program Engineer performs a “Functional Failure Evaluation”, as per DPT-PE-00008, against these components on a functional basis (i.e., how will failure of the component affect system performance?). This analysis captures the degradation mechanisms, as well as the remedial actions.

Table 6 presents a snapshot of ageing-related performance indicator results for Tier 1 systems, as extracted in December 2014 from the System Health Reports in System IQ. The Performance Indicators identified in Table 6 demonstrate that ageing-related indicators are included in performance monitoring. Additional performance indicators are monitored, trended and analysed.

If monitoring/trending indicates that the SSC performance has degraded then the Responsible System Engineer/Station Component Engineer/Program Engineer 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 [53], Equipment Reliability Problem Identification and Resolution).

If results from monitoring/trending activities have identified a degraded SSC condition, the Responsible System Engineer/Station Component Engineer determines if an SCR is required in accordance with the guidance provided in BP-PROC-00782 [53].

With respect to safety related structures, NK21-PIP-20000-00001, CSA N291 In-Service Inspection Program for Bruce NGS A Safety Related Structures, 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 [20] is listed as condition 1.7 in the PROL [15], 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[8], 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.

[8]Reporting is performed under S-99 up to the end of 2014, and under CNSC REGDOC-3.1.1 for periods thereafter.

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 (U1/U2/U3/U4)		Maintenance backlog (U1/U2/U3/U4)					Operational Challenges (U1/U2/U3/U4)		ER Clock Resets
	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/R/R/R	G/G/G/G	Y/R/R/Y	G/G/G/Y	G/W/G/G	Y/G/G/G	W/R/R/W	G/G/G/G	R/R/R/G	G/G/G/G
SDS2	R/R/R/R	W/G/G/G	R/Y/R/R	W/G/W/Y	W/G/R/Y	G/G/G/R	R/R/R/R	Y/Y/G/G	R/R/R/G	W/G/G/G
Negative Pressure Containment	G/G/G/G	G/G/G/G	G/G/G/G	Y/Y/G/R	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	Y/G/W/G	G/G/W/G	Y/R/G/R	W/W/R/Y	W/Y/G/R	W/G/R/R	R/R/R/G	G/G/G/G	G/G/G/G	G/G/G/G
EFADS & PARMS UNIT 0A	Y	W	R	G	G	G	R	G	G	G
Feed, Bleed, Relief and Storage Recovery	G/G/Y/W	G/G/G/G	R/W/G/R	W/W/W/Y	R/G/G/R	Y/G/Y/R	R/R/R/R	Y/Y/G/G	G/G/G/G	G/G/G/G
Emergency Coolant Injection UNIT 0A	G	G	R	W	R	R	R	R	G	G
Emergency Coolant Injection – UNITS 1,2,3,4	G/G/G/G	G/G/W/W	Y/G/Y/R	G/G/W/W	W/G/G/W	G/G/G/W	R/R/R/R	G/G/Y/R	Y/G/Y/G	G/G/G/G
Service Water	G/G/G/G	G/G/G/G	R/R/R/R	Y/Y/Y/R	R/R/R/R	R/R/R/R	R/R/R/R	G/G/G/G	G/G/Y/G	G/G/W/G
Instrument and Service Air	G/G/G/G	G/Y/Y/W	R/R/W/W	R/R/R/W	R/R/R/Y	R/R/R/R	R/R/R/R	G/G/G/G	G/G/G/G	-
Powerhouse Emergency Venting	G/G/G/G	G/G/G/G	R/R/R/R	G/G/G/G	R/R/R/R	G/G/G/G	G/G/G/G	G/G/G/G	G/G/G/G	-

Performance Indicator	Functional Failures (U1/U2/U3/U4)		Maintenance backlog (U1/U2/U3/U4)					Operational Challenges (U1/U2/U3/U4)		ER Clock Resets
	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	
Powerhouse Air Conditioning	R/W/R/G	G/W/G/G	W/W/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	G/G/G/G
Control Room and Chilled Water Systems UNIT 0A	R	W	R	-	R	-	R	G	G	Y
Main Heat Transport Circuit, Gland Seal Circuit, Feeders, Autoclaves	W/W/G/G	G/G/G/G	G/W/W/Y	W/R/R/R	W/W/G/R	R/R/Y/R	R/R/R/R	Y/Y/G/G	G/Y/G/Y	R/R/G/R
Emergency Boiler Cooling	G/R/G/G	G/G/W/W	G/G/G/R	G/G/G/G	G/G/W/G	G/R/W/R	R/G/W/R	G/G/G/G	G/G/Y/G	-
Class III	G/G/Y/Y	G/G/G/G	W/G/W/W	G/G/W/G	G/Y/R/R	W/G/W/W	G/G/R/R	G/G/G/G	G/G/G/G	G/G/G/G
Class IV	G/G/G/G	G/G/G/G	W/W/Y/G	G/W/Y/Y	R/W/R/R	W/W/R/W	G/G/G/R	G/G/G/G	G/G/G/G	G/G/G/G
Standby Generators	G/Y/R/W	G/G/G/G	R/R/G/R	G/G/W/G	G/G/G/W	G/G/G/G	R/R/R/R	Y/Y/Y/Y	G/G/G/G	G/G/G/G

Review Task Conclusion

The process described in BP-PROC-00781 [52] 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 [51], BP-PROC-00781, Performance Monitoring [52], and BP-PROC-00782, ER Problem Identification and Resolution [53], 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 [118] to ensure that the preparation, distribution and maintenance of documents is 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 [119], defines the process of managing the life cycle of Bruce Power Controlled Documents and BP-PROC-00098, Records Management [120], 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 [121], 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 [122]. 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 [123], describes the process for management and administration of PASSPORT and related systems, including Livelink, 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 Preventive Maintenance Implementation, BP-PROC-00780 [51], Performance Monitoring, BP-PROC-00781 [52], and Equipment Reliability Problem Identification and Resolution, BP-PROC-00782 [53], which are interfacing procedures that lead the continuous improvement process.

BP-PROC-00780, Preventive Maintenance Implementation [51], 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) 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 [52], 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 & Identification of Critical SSCs [49].

Bruce A 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 [52]. A table of components and programs scoped into the performance monitoring program has been included as Appendix C to BP-PROC-00781 [52]. 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 [52] 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 this procedure. Reviews of the lists are conducted in accordance with the requirements of BP-PROC-00068 [119] which governs the life-cycle of controlled documents.

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 [52] 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 Livelink, Meridium and E-suite, for example:

- Preventive Maintenance results (as per BP-PROC-00284 [64], 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 Livelink).
- Inspection results from PASSPORT.
- As Found Condition reports from the As Found Condition Coordinator (AFCC) (“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.).

BP-PROC-00782, Equipment Reliability Problem Identification and Resolution [53], describes the problem resolution process, including the interface with the Station Condition Record (SCR) Process (BP-PROC-00060 [74]) and the Action Tracking Process (BP-PROC-00019 [75]). 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. Required Corrective Maintenance is executed according to the procedures under BP-PROG-11.04, Plant Maintenance Program [76].

SCRs are stored in and can be retrieved from E-Suite, and TOEs are available from Livelink. 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 [20] is listed as condition 1.7 in the PROL [15], 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 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 [124]. NS-G-2.12 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 [39], is based in part on NS-G-2.12, 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, with an emphasis on the need for a proactive and integrated methodology.

Review Task Assessment

CNSC REGDOC-2.6.3 [39] supersedes RD-334 [40], 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 [40] near the end of 2012 and submitted it to the CNSC in NK21-CORR-00531-10027/NK29-CORR-00531-10447 [114]. In this assessment a number of gaps were identified, and the CNSC requested in NK21-CORR-00531-10278 [125] that Bruce Power provide more definitive information for the residual Risk Control Measures (RCMs) relating to RD-334 [40] and the closing of identified gaps by providing specific actions and timelines. This information was subsequently provided in NK21-CORR-00531-10336 [126] and is summarized below:

(a) RD-334 [40], Clause 3.1

- Gap: FORM-10700, Design Scoping Checklist [127] does not include service life or ageing.

- Response: As stated in RD-334 [40]: "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 [40] asks for mitigating strategies to be added to design documentation to include design features that mitigate the effects of ageing mechanisms. RD-334 [40] 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 [127] from BP-PROC-00539 [128]) will be updated to reflect RD-334 [40] Section 3.1.
- Status: This action has been completed per Action Request REGM 28332951-06.

(b) RD-334 [40], 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 [71], per Action Request REGM 28332951-05.

(c) RD-334 [40], Clause 3.4.3

- Gap: Possibility of extended shutdown should be included in BP-PROC-00400, Life Cycle Management Plans for Critical SSCs [70].
- Response: RD-334 [40] 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 [70] 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 [40], 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 [129], 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, [59] and BP-PROC-00400 [70].

- Status: This action has been completed per Action Request REGM 28332951-09.

(e) RD-334 [40], Clause 4.4.1, Item 4

- Gap: Anticipated obsolescence issues: BP-PROC-00533, Obsolescence Management [56], 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 [47].
- 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 [56] and BP-PROC-00532, Critical Spare Parts and Strategic Component Assessment [58], with the appropriate governing documents (e.g., BP- PROC-00783 [47] and BP-PROC-00778, Scoping and Identification of Critical SSCs [49]). This will be done via an update to BP-PROG-11.01, Equipment Reliability [46], Appendix A. BP-PROG-11.01 [46] is currently being reviewed and the implementing procedures (including BP-PROC-00783 [47]) will be updated to include reference to the new BP-PROC-00533 [56].
- Status: This action has been completed per Action Request REGM 28332951-10.

(f) RD-334 [40], Clause 4.4.1, Item 7

- Gap: RD-334 [40] 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 [40] 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 [40] was included in the 2013 interim PSR, and documented in Appendix N of the 2013 Safety Basis Report [2]. In April 2014, CNSC staff re-classified CANDU Safety Issue GL3 “Ageing of Equipment and Structures” from Category 3 to Category 2 [117].

Bruce Power submitted a plan for transition to CNSC REGDOC-2.6.3, which supersedes RD-334, in December 2014 [40].

Review Task Conclusion

Given that all the actions identified above have been closed, and considering Bruce Power’s plan for transitioning to CNSC REGDOC-2.6.3 implementation [41], it is concluded that ageing management at Bruce Power is in compliance with CNSC REGDOC-2.6.3 [39]. 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.[9]

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 [46], and its implementing procedures, i.e., Scoping and Identification of Critical Components BP-PROC-00778 [49], Continuing Equipment Reliability Improvement BP-PROC-00779 [50], Preventive Maintenance Implementation BP-PROC-00780 [51], Performance Monitoring BP-PROC-00781 [52], and Equipment Reliability Problem Identification and Resolution BP-PROC-00782 [53], 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
- 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 [93], and Fuel Channel Condition Assessment (FCCA), B-REP-31100-00003 [130], 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 crack initiation due to in-service

[9] BP-PROC-00786, "Margin Management" [131], describes how Bruce Power manages design and operating margins.

induced flaws and material wear due to debris, corrosion, erosion and the passage of fuel bundles. The FCLCMP and FCCA also illustrates the interactions between the various degradation mechanisms, and the FCLCMP addresses the requirements of CSA-N285.4 [22] through the Fuel Channel Periodic Inspection Program.

The FCLCMP and Fitness for Service assessments implement both deterministic and probabilistic approaches as permitted in CSA N285.8-05 [132] and subject to regulatory approval. CSA N285.8-10 [42] provides guidance on probabilistic evaluation of pressure tube degradation mechanisms. This is highlighted in the code-to-code comparison of CSA-N285.8-10 [42] with the previous version of this standard, CSA-N285.8-05 [132], that 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).

In the latest pressure tube flaw assessment for the Bruce A Unit 3 2014 planned outage (NK21-CORR-00531-11657 [133] sent to the CNSC November 2014) Bruce Power is reporting compliance with Clause 8.2 of CSA-N285.8-10 [42], but has not been required to report compliance with Clauses 8.3, 8.4, and 8.5 as there was no material surveillance examination done in the Bruce A Unit 3 2014 planned outage.

The acceptance criteria in Clause 8 of CSA-N285.8-10 [42] continue to improve as new results from COG R&D activities become available. The latest update to the CNSC on this topic was provided in NK21-CORR-00531-11339 [134]. In this reference, Bruce Power states:

"The strategic objectives for the fuel channel R&D program (COG-12-9106) for 2013 – 2018 are:

- provide sufficient confidence and data to support the application of the CSA Standard N285.8 for pressure tube fitness for service, and subsequent support for updating as necessary;"

Hence, through active participation in joint projects of the CANDU Owners Group on fuel channel integrity and participation in the update of the CSA standard, Bruce Power is in the forefront of knowledge related to fuel channel ageing mechanisms and is compliant to

CSA N285.8-10 [42] as outlined in the report (B-REP-31100-00010 [43]) 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 [93]. In summary, 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 crack initiation due to in-service induced flaws and material wear due to debris, corrosion, erosion and the passage of fuel bundles.

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 [135]. This procedure documents the process for managing the condition of PHT feeder piping.

A detailed account of ageing degradation mechanisms for the PHT Feeder Piping is provided in the Life Cycle Management Plan, B-LCM-33126-00001 [95] that was prepared in accordance with the procedure for the Life Cycle Management of Critical SSCs, BP-PROC-00400 [70], and complies with all requirements of CSA N285.4-05 [136].

The PHT Feeder Piping LCMP, B-LCM-33126-00001 [95] 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 [137].
- A description of research and development 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.

The LCMP also includes additional information regarding Unit 1 and 2 restart details, West Shift Plus, feeder ice plugging information, leakage monitoring, historical inspection findings, and inspection schedule information.

As documented in the PHT Feeder Piping Life Cycle Management Plan, B-LCM-33126-00001 [95], 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. Flow Accelerated Corrosion 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 [96]. 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 [96] 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 [96] satisfies the requirements of the Bruce Power Procedure BP-PROC-00267, Management of Steam Generator and Preheater Tube Integrity. It has been updated to include Bruce Units 1&2 replacement steam generators and original preheaters.

Overall, the degradation mechanisms affecting steam generators and pre-heaters have been identified and assessed, and mitigating actions have been developed, as documented in the Steam Generator and Preheater Life Cycle Management Plan, B-PLAN-33110-00001 [96]. The greatest challenge is due 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 [55].
- 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 a list of 36 SSCs that meet the above criteria and for which LCMPs either exist or are under development. This list was prepared using information from BP-PROC-00400 [70] augmented by information from a PASSPORT search conducted in October 2013 as part of Appendix F of the 2013 Safety Basis Report [2], and updated in December 2014. A separate list of 19 SSCs for which LCMPs will be prepared is also provided in Table 8. In 2014, four of the existing LCMPs were revised (Calandria Shield Tank Assembly, Heat Exchanger and Condenser, PHT Feeder Piping and Large Transformers) and one new LCMP was issued (Service Water Piping).

Table 7: List of Existing (or In Progress) LCMPs for Bruce A and Bruce B

SSC	LCMP Document Number	Document Revision	Document Date
Cables	BP-PLAN-00044	R000	21DEC2010
Calandria Shield Tank Assembly	B-LCM-31200-00001	R000	29SEP2014
Civil Structures	B-PLAN-20000-00001	R000	07JUL2010
Computer Systems	BP-PLAN-00057	R000	12MAR2013
ECI Recovery Heat Exchangers	B-PLAN-34340-00001	R000	09JAN2009
Electrical Systems	BP-PLAN-00045	R000	28MAR2011
End Shield Cooling Heat Exchangers	B-PLAN-34110-00001	R000	09JAN2009
Feeder Piping	B-LCM-33126-00001	R000	29SEP2014
Fuel Channels	B-PLAN-31100-00001	R005	26OCT2012
Generators and Auxiliaries (BA)	BP-PLAN-00042	R001	31OCT2012
Generators and Auxiliaries (BB)	BP-PLAN-00043	R000	15NOV2010
Heat Exchangers and Condensers	B-LCM-04660-00001	R000	04Apr2014
Irradiated Fuel Bay Heat Exchangers	B-PLAN-34400-00001	R000	03JUL2008
Instrumentation & Control	BP-PLAN-00047	R000	21JUN2011
Large Pumps	BP-PLAN-00053	R000	07OCT2011

SSC	LCMP Document Number	Document Revision	Document Date
Large Pump Motors	B-PLAN-05600-00001	R000	30AUG2010
Large Transformers	B-LCM-50000-00001	R000	02DEC2014
Main Condenser	B-PLAN-42110-00001	R001	28FEB2013
Maintenance Cooling Heat Exchanger	B-PLAN-34720-00001	R000	09JAN2009
Moderator Heat Exchanger	B-PLAN-32110-00001	R000	09JAN2009
Moderator Purification System Heat Exchanger	B-PLAN-32210-00001	R000	09JAN2009
Negative Pressure Containment System Components	B-PLAN-34200-00001	R000	18DEC2013
Nuclear Piping	BP-PLAN-00052	R001	08AUG2013
Pressure Vessels & Tanks	B-PLAN-04600-00001	R000	15NOV2013
Printed Circuit Boards	B-PLAN-60000-00002	R000	24AUG2013
Reactivity Control Units (BA)	BP-PLAN-00051	R000	19OCT2011
Reactivity Control Units (BB)	BP-PLAN-00059	R000	22OCT2012
Service Water Piping	B-LCM-07211-00001	R000	07Oct2014
SDC Heat Exchanger	B-PLAN-34710-00001	R000	IN PROGRESS
Small Pumps and Motors	B-PLAN-04610-00001	R000	01JAN2013
Steam Generators & Preheaters	B-PLAN-33110-00001	R004	18FEB2011
Traveling Screens & Trash Bar Screens	B-PLAN-71120-00001	R000	07AUG2013
Turbines & Auxiliaries (BA)	BP-PLAN-00040	R000	21DEC2010
Turbines & Auxiliaries (BB)	BP-PLAN-00041	R000	15NOV2010
Vacuum Building Fibreglass Reinforced Plastic Spray Header (BA)	NK21-PLAN-34320-00003	R001	27SEP2011
Valves	BP-PLAN-00037	R000	20DEC2010

Table 8: List of Planned New or Revised LCMPs

SSC	Existing LCMP?
Air Operated Valves	No
Air Systems	No
Battery Banks	No
Buried Piping	No
Buses, Switches and Switchgear	No
Cables	Yes
Circuit Breakers	No
Cranes	No
Buildings & Structures	Yes
Computers	Yes
Emergency Power Generators	No
Emergency Power Systems and Qualified Power Systems	No
Fuel Channels	Yes
Fuel Route	No
Generators & Auxiliaries	Yes
Instrumentation & Control	Yes
Inverters, Converters, Rectifiers in Class I and II Electrical Power	No
Isolated Phase Bus	No
Manual Valves	No
Motors	Yes
Motor Control Centres	No
Motor Operated Valves	No
Non-Return Valves	No
Nuclear Piping	Yes
PHT & Moderator Upgraders	No
Pressure Relief Devices	No
Printed Circuit Boards	Yes
Pumps	Yes

SSC	Existing LCMP?
Reactivity Control Units	Yes
Secondary Piping	No

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 [138] to demonstrate safe operation of the Bruce A units under 2019 aged conditions.

Review Task Conclusion

The focus of the assessment was on the dominant ageing mechanisms of critical SSCs, and 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. Although only 3 major LCMPs were discussed more than 36 similar plans exist or are under development while another 19 are planned. 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 [50], 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 [49] 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 [59], 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 [51] and also captured in BP-PROC-00783, Long Term Planning and Life Cycle Management [47].

Documenting the equipment as found condition is important to a continuously improving equipment reliability process, and BP-PROC-000780 [51] presents the process for capturing information from maintenance personnel on the as-found condition and providing feedback to the Responsible System Engineer/Station Component Engineer.

BP-PROC-00781 [52] 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.

- Monitoring by Responsible System Engineers/Station Component Engineers walkdowns.

The performance criteria and monitoring parameters are obtained from the SPMPs/CPMPs (DPT-PE-00008 [65]) or TBAs prepared in accordance with BP-PROC-00779 [50]. Performance monitoring results are recorded in System Health Reports (SHR) or Component Health Reports (CHR) as per the intervals established in the System Health Reporting procedure (DPT-PE-00010 [66]) or Component Health Reporting procedure (DPT-PE-00011 [67]).

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, BP-PROC-00400 [70]. Inaugural/baseline inspection data are collected in compliance with the requirements of CSA N285.4 and N285.5, as described in Periodic Inspection, BP-PROC-00334 [87]. Data from periodic inspections are also collected, and findings are reviewed, evaluated and dispositioned.

For safety related structures, NK21-PIP-20000-00001, "CSA N291 In-Service Inspection Program for Bruce NGS A Safety Related Structures" 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 12.

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 heat transport system (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 have been addressed for parameters that are measured continuously at Bruce Power; explicitly through compliance processes, such as corrections to detector calibration factor to address high reactor inlet temperatures and through physical plant changes, such as Steam Generator (SG) ID 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 Safe Operating Envelope (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 are 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 [71], Integrated Aging Management for Safety Assessment, 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 [39] (which supersedes RD-334 [40]).

Execution of DPT-NSAS-00016 [71] 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 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 SF5 Report, Deterministic Safety Analysis. Specifically, Section 5.3 of the SF5 Report presents an evaluation of the existing safety analysis and validity of assumptions given actual conditions of the plant.

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 A Operating Policies and Principles [139] outline operating boundaries within which the Bruce A 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 [72] 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 [77]), performance monitoring with respect to chemistry control (DPT-CHM-00007 [78]) and the outage chemistry program (DPT-CHM-00008 [79]). 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). To remain below the RIHT limits, Units 3 and 4 are operated at a lower reactor power with some lowering of steam generator pressure.

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

Effective Preventive Maintenance Implementation [51] and Performance Monitoring [52] Programs, supported by periodic and in-service inspection and testing programs [87], are used to continuously confirm effectiveness of monitoring and mitigation ageing.

BP-PROC-00780, Preventive Maintenance Implementation [51], 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 [61], 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 [52], provides the basis and expectations for the Equipment Performance Monitoring Process. Performance Monitoring is supported by BP-PROC-00284, Predictive Maintenance (PdM) [64] 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.

Heat Transport System 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
- West Shift Plus to address pressure tube axial elongation due to irradiation
- Selective defuelling of fuel channels to reduce deformation
- Implementation of modified 37-elment (37M) fuel to mitigate the impact of HTS ageing on margins to critical channel power
- 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, includes 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 the report on SF2, 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 [60], 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 [46]). The scope of SSCs to be included in the condition assessment process is identified through the LCM process (BP-PROC-00400 [70]), 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 Performance Monitoring (BP-PROC-00781 [52]).

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 [140], governs the management of materials in storage, and the implementing procedure is BP-PROC-00262, Warehouse Operations [141]. Section 4.3.2 of this procedure reads "Items and materials with limited shelf life are identified on the Cat ID as established by PE in SEC-PE-00015, Selection of Item Shelf Life Requirements [142]. Storage and monitoring requirements are implemented and maintained by the FLM, Warehouse - Stockkeeping in accordance with DPT-MM-00007, Item Shelf Life Management [143]."

Materials that have been deployed from stores and that deteriorate while being used fall under BP-PROG-11.04 Plant Maintenance [76], under the category of preventive maintenance (BP-PROC-00501 [61]). Longer term degradation not addressed by routine preventive maintenance is addressed by ageing management (BP-PROC-00400 [70]).

Within BP-PROC-00501 [61], there is very little direct guidance on how to establish periodic maintenance to address degradation of materials. Similarly, BP-PROC-00400 [70] 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 [144] invokes BP-PROC-00135, Station Rework Program [145].

The Steam Generator and Preheater Life Cycle Management Plan, B-PLAN-33110-00001 [96] 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
- 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. Other properties are monitored through the Fuel Channel R&D program sponsored by the CANDU Owners Group.

CNSC staff has indicated that the 2009 version of CSA N285.4 with the 2011 Update will be included in the next PROL [33]. 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 12.

As documented in the Feeder Piping LCMP, B-LCM-33126-00001 [95], 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.

A Life Cycle Management Plan [146] has been developed for monitoring any degradation in the material properties of the Fiberglass Reinforced Plastic Spray Headers in the Bruce A Vacuum Building, particularly in the context of long term operation of any refurbished units. In addition, the periodic inspection program for the vacuum building [147] includes external inspection of the silicone rubber roof seal between the perimeter wall and the roof slab, for possible degradation and ageing.

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 12.

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

In the "IAEA Safety Glossary - Terminology Used in Nuclear Safety and Radiation Protection 2007 Edition", the definition for technological obsolescence redirects the reader to the definition of non-physical ageing. The definitions of "ageing" and "non-physical ageing" from the IAEA Safety Glossary are described below:

> *"Ageing. General process in which characteristics of a structure, system or component gradually change with time or use.*
>
> - *Although the term ageing is defined in a neutral sense — the changes involved in ageing may have no effect on protection or safety, or could even have a beneficial effect — it is most commonly used with a connotation of changes that are (or could be) detrimental to protection and safety (i.e. as a synonym of ageing degradation).*
>
> *Non-physical ageing. The process of becoming out of date (i.e. obsolete) owing to the evolution of knowledge and technology and associated changes in codes and standards.*

- *Examples of non-physical ageing effects include the lack of an effective containment or emergency core cooling system, the lack of safety design features (such as diversity, separation or redundancy), the unavailability of qualified spare parts for old equipment, incompatibility between old and new equipment, and outdated procedures or documentation (e.g. which thus do not comply with current regulations).*
- *Strictly, this is not always ageing as defined above, because it is sometimes not due to changes in the structure, system or component itself. Nevertheless, the effects on protection and safety, and the solutions that need to be adopted, are often very similar to those for physical ageing.*
- *The term technological obsolescence is also used."*

These aspects of non-physical ageing are covered in Bruce Power's governance.

Regarding "unavailability of qualified spare parts for old equipment", this is covered by the Obsolescence Management procedure, BP-PROC-00533 [56]. Obsolescence management requires the development of an action plan that maximizes the benefits of equipment modernization and results in a cost-effective solution. Action Requests (type OBSE - obsolescence action plan) are created to track and document the Action Plans and resolution of obsolescence issues emanating as a result of identification during the performance of normal work tasks, an audit, or assessment. For all obsolescence action plans created, Bruce Power reviews nine possible solution paths including substitution/equivalency, rebuild / repair, or design changes, each of which includes a review of technological options. The associated 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. This procedure aligns with INPO AP 913 and other EPRI guidance.

The other aspects of non-physical ageing effects are dealt with by the current design management and equipment reliability programs, as well as PSR/ISR processes.

An example of Bruce Power's managing obsolescence of technology is consideration of transition from analog technology to digital technology in I&C aspects of the plant when feasible.

Review Task Conclusion

Bruce Power's governance conforms with obsolescence of technology.

6. Interfaces with Other Safety Factors

There is some degree of interrelationship among most of the 15 Safety Factors that comprise the Bruce A ISR. 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 in Section 5.14 the assessment of the verification of the actual state of SSCs against the design basis. “Safety Factor 2: Actual Conditions of SSCs” Section 5.9 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 provide a cross section, intended to 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 the assessment are compared with business needs, the Bruce Power management system, industry standards of excellence and regulatory/statutory or other legal requirements.

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 2008 are listed in Table 9 as evidence that program effectiveness is being monitored.

Table 9: Self Assessments Relevant to SFR4 Conducted Since 2008

Assessment Number	Title
SA-BAOP-2008-04	Conduct of Equipment Alignment Checks
SA-BAOP-2008-08	Conduct of Operator Field Inspections – Implementation Effectiveness
SA-CHM-2008-04	Chemistry Performance Reporting
SA-MP-2008-04	Outage Readiness
SA-MPA-2008-03	Corrective Maintenance Backlog
SA-MPR-2008-01	FME - Site Program Compliance
SA-MS-2008-01	Assessing Effectiveness for Innage/Outage
SA-MS-2008-03	Standard adherence of Canada CAN/CSA-ISO 9001
SA-OTG-2008-01	Carry out review of forced outage plans and process
SA-OTG-2008-03	Review Online to Outage Handovers for B841
SA-OTG-2008-04	Review Outage Milestone Compliance
SA-PE-2008-01	Use of trending in System Performance Program
SA-PRPM-2008-01	Opportunity to improve building maintenance processes
SA-RS-2008-03	Maintenance of NOP operating margin and effectiveness
SA-SSSB-2008-03	Safety Inspections
SA-CLA-2009-01	Outage Chemistry
SA-CLA-2009-02	Online Instrumentation QA/QC/Preventive Maintenance
SA-COM-2009-01	Component Design Basis Inspection
SA-ELCE-2009-09	PdM Program Assessment
SA-ELCE-2009-10	Preventive Maintenance Technical Basis Assessment
SA-MPR-2009-02	Foreign Material Exclusion
SA-MS-2009-08	Status Control of Mock-ups

Assessment Number	Title
SA-PDE-2009-02	Independent Assessment of Bruce 1 and 2 Restart Eng Delegate Performance
SA-PE-2009-03	System Performance Monitoring – Walk Downs
SA-RS-2009-02	Corrective action planning effectiveness
SA-WM-2009-02	BA/BB Oversight & Reduction of Corrective Mtce, Elective Mtce & High Priority Backlogs
SA-WM-2009-03	BA and BB Outage Scope Selection and Scope Stability
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
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

Assessment Number	Title
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-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
SA-ERI-2014-07	Quality of System Health Reporting
SA-MPR-2014-08	SECNMMM Equipment Capability

A subset of the self-assessments listed in Table 11 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" [129]

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 Rev. 1 (now at Rev. 2 [70]). Moving forward, it will be necessary to begin revising these LCMPs in order to bring them into compliance with Rev. 1 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/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's listed as an objective on the 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 BA and two for BB) 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” [148]

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 prior to 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 (EWMS).

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.

<u>SA-ERI-2014-01 "Review of Data Needs to Assess SSC Aging"</u> [149]

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).

No adverse conditions were identified during this FASA, and one opportunity for improvement was identified, 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. The assignment associated with this SCR was scheduled to be completed by the end of November 2014.

<u>SA-ERI-2014-07 "Quality of System Health Reporting" [150]</u>

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. As a result of this SCR, DPT-PE-00010 "System Health Reporting" and BP-PROC-00559 "Station Plant Health Committee" will be 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" will be revised.

<u>SA-MPR-2014-08 "Quality of System Health Reporting" [151]</u>

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, 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 is scheduled to be completed March 30, 2015.

"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:

"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 its difficult to release staff to benchmark and find appropriate places to visit." This is being addressed by SCR 28451474 (due date June 20, 2015).

"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).

7.2. Internal and External Audits and Reviews

The objective of the audit process as stated in BP-PROG-15.01 [152] 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 [153]).

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 Performed Before 2008

At the time of the 2008 ISR, although the Equipment Life Cycle Engineering (ELCE) Department (now called the Component and Program Engineering (CAPE) Department) had not been audited recently, several audits had been performed in the two years prior to 2008 on functions that were closely related to ageing management. Of particular note was an audit conducted in April 2006 by Bruce Power's Quality Assurance Department on the implementation of BP-PROC-00334 [87] on Periodic Inspection. The results of this audit are documented in AU-2006-00015 [154]. This audit concluded that periodic inspections identified, planned and conducted met the intent of BP-PROC-00334 [87]. However, some issues with respect to the program definition and implementation were identified for further attention.

7.2.2. External Audits and Reviews Performed Before 2008

To support the return to service of Bruce 1&2, Bruce Power had hired an independent consultant to perform a systematic review of safety based on IAEA NS-G-2.10 [155]. Table 4.1.4-1 in the resultant Bruce 1&2 ISR Report [9] suggested improvements related to ageing. The related elements and their status at the time of the 2008 ISR are provided in Table 10.

In addition, the CNSC had provided a number of comments on this Safety Factor for the Bruce 1&2 ISR review, as documented in Section 5.27 of Reference [156]. The shortcomings arising from these comments were the following:

- Bruce Power should review their ageing management program governing documents against the guidance documents identified in paragraph 4.24 of IAEA NS-G-2.10 [155];
- The governing program documents had yet to be produced and as such, a detailed review of Bruce Power's submission by CNSC staff could not be performed; and
- CNSC staff requested Bruce Power to provide a progress update on processes and plans for development of a systematic and integrated Ageing Management Program.

Bruce Power had subsequently responded to the CNSC comments by way of Reference [157]. At the time of the 2008 ISR, Bruce Power had conducted the initial steps necessary for ageing management, which was deemed consistent with a modern version of a risk-informed approach to ensuring that plant SSCs achieve the desired target life and also places Bruce Power in a position to technically support the potential for plant life extension.

Elements of this program included component/system SPV, ageing, obsolescence, critical spares, and where required the development of life-cycle plan(s) and strategies were being initiated. Alternatively, in the case where a full life-cycle plan and strategy was not deemed appropriate, a condition assessment for components/systems was being completed. Table 10 provides a status update from the 2008 ISR.

Table 10: Bruce 1&2 ISR Report Suggested Improvements Related to Ageing

Observation	B1&2 Comment	Status in 2008 Review
Establish an integrated Ageing Management program for Bruce 1&2	Bruce Power had committed to establishing an ageing management program prior to the return to service of Bruce 1&2	At the time of the 2008 review, Bruce Power was in the process of establishing the Ageing Management Program.
Establish CSA N285.4, N285.5 & N287.7 Inspection Programs	These inspection programs had been credited in the Safety Factors for plant design and actual condition of SSCs. These are one of several components that were being incorporated into the ageing management program.	Bruce Power Periodic Inspection Programs to meet CSA N285.4, N285.5, and N287.7 were described in the 2008 review.

7.2.3. Audits Performed Since 2008

This assessment reviewed the relevant internal audits that were conducted in the five years since 2008. 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 11.

Table 11: Audits Relevant to SFR4 Conducted Since 2008

Audit Number	Title
Plant Reliability Integration	
AU-2008-00010	Preventive Maintenance Program
AU-2008-00012	Switchyard Maintenance
AU-2008-00027	Relief Valve Program

Audit Number	Title
AU-2009-00010	BRUCE A VBO Periodic Inspection
AU-2009-00034	RV field Repairs
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
Inage Work Management	
AU-2010-00022	H1/H2 Work Prioritization
AU-2012-00014	On-line Work Management
Outage Work Management	
AU-2009-00035	Outage Milestones
AU-2009-00043	Outage Management
AU-2010-00026	Forced Outage Management
AU-2013-00008	Outage Management
Plant Maintenance	
AU-2009-00003	CMLF ISO 9001 Program
AU-2009-00031	Corrective Maintenance Backlog
AU-2009-00032	Turbine Crew FME practices
AU-2010-00008	ISO 9001
AU-2010-00012	GSB Task Planning
AU-2011-00027	Foreign Material Exclusion
AU-2013-00006	Maintenance Program

Audit Number	Title
Chemistry Management	
AU-2008-00048	Plant Status Control Assessments
AU-2011-00024	Chemistry Management Program
AU-2011-00026	Outage Chemistry Program
AU-2014-00010	Control of System Chemistry

Audits AU-2013-00006 and AU-2014-00010, which were performed more recently and are more directly relevant to aging, are summarized below.

AU-2013-00006, Maintenance Program [158]

The objective of this audit was to evaluate whether BP-PROG-11.04 Plant Maintenance [76] 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 requires that Bruce Power implement and maintain a maintenance program in accordance with CNSC regulatory document S-210 Maintenance Programs for Nuclear Power Plants [30]. BP-PROG-11.04 Plant Maintenance [76] 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 [76] 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 [76] that satisfy the S-210 requirements [30].

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-11.04to 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.

<u>AU-2014-00010, Control of System Chemistry</u> [159]

This audit evaluated the effectiveness of, and compliance to, DPT-CHM-00003 R006, Control of Chemistry [77]. 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 [77] 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.
- SCRs 28439136, 28439262, 28439135: Chemistry Program Requirements are not adequate or complete.
- SCRs 28439139, 28439264, 28439137: Audit (AU-201 1-00024) and FASA (SA-CHM-2012-01) Corrective Actions found ineffective.
- SCRs 28439141, 28439265, 28439140: Control of Chemistry Program records not adequately controlled or maintained.
- SCRs 28439143, 28439273, 28439142: Chemistry staff qualifications are not adequately established.
- SCR 28439144: Control of Chemistry Program Non-Compliance to BPMS Procedural requirements.

Opportunities for Improvement:

- SCR 28439145: Control of Chemistry Program Description in BP-PROG-1 2.02 requires updating.
- SCR 28439146: Control of Chemistry SCA Trending expectations should be clearly documented and aligned.
- SCR 28439147: EPRI Strategic Water Chemistry Plans should be established.

The assignments associated with these SCRs are either complete or in progress and scheduled to be completed in February 2015 or after.

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 operators 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.

7.3.1. Regulatory Evaluations and Reviews Performed Before 2008

At the time of the 2008 review, there were no recent regulatory evaluations and reviews that addressed ageing exclusively. The CNSC's evaluation of the Bruce A maintenance, inspection and reliability programs in its report for 2007 on the Safety Performance of the Canadian Nuclear Power Industry [160], was however relevant to ageing.

Maintenance

The CNSC noted in this report that Bruce Power had policies, processes and procedures in place providing direction and support for its maintenance program, and that the program met CNSC expectations. It noted, however, that a tightening of the definition of Corrective Maintenance in late 2007 had caused a one-time increase in the number of maintenance items, which had the consequence of more than doubling the backlog indicator numbers at both Bruce A and B. Bruce Power was in the process of submitting an action plan showing how the maintenance backlog would be brought in line with the existing Bruce A and B targets by the end of May 2008. As requested by CNSC staff, Bruce Power submitted a response indicating that Bruce A and B were still in compliance with the licence conditions on maintenance [161]. "Safety Factor 8: Safety Performance" had identified maintenance backlog as a key issue and had included proposed corrective actions.

Reliability

The 2007 CNSC Report [160] also noted that Bruce Power informally submitted a partial set of documents to support the reliability program. Bruce Power had been requested to re-submit for review their complete reliability program by June 2008, as part of licence renewal. All Systems Important to Safety at Bruce 3&4 met the reliability targets, with the exception of SDS2, which did not meet the target due to an Environmental Qualification issue. CNSC staff indicated that the issue had been addressed, and would not have an impact on the future reliability of the systems [160]. Bruce Power's Performance and Condition Monitoring procedure was revised in March 2008 and the revision included an update to the systems list to have a better alignment/consistency with the S-98 report.

Periodic Inspection

The 2008 Bruce A licence [162] specifically referenced the requirements for periodic inspection. In 2005, Bruce Power submitted an extensive review of Bruce 3&4 inspections in support of compliance with N285.4 and N285.5 [163]. The review determined that some inspections were required and Bruce Power provided a plan and schedule to complete the identified inspections.

A compliance evaluation of N287.7-96 was completed as part of the Bruce 1&2 ISR to address a category two CNSC issue. In that submission, Bruce Power committed to revising the Periodic Inspection Plan for concrete containment structures and submitting it to the CNSC prior to the containment outage inspection required by licence condition 5.2(c). The current periodic inspection plans for concrete containment structures are documented in NK21-PIP-21100-00001 "CSA N287.7-08 Periodic Inspection Program for Bruce NGS A Concrete Containment Structures and Appurtenances (Excluding Vacuum Building)" [164] and NK21-PIP-25100-00001 R002, "CSA N287.7-08 Periodic Inspection Program for Bruce NGS A Vacuum Building" [147].

The 2007 CNSC Report [160] indicated that, in the area of Structural Integrity, CNSC staff was satisfied both with the inspection work and the licensee's assessment of inspection findings at Bruce A.

7.3.1.1. CNSC Issues from B1&2 Review

CNSC staff raised a number of issues following the review of Bruce 1&2 return to service submissions [156] of which Issue 5.27.1, Program Documents Review Against Current Standards, Issue 5.27.2, Submission of Program Documents, and Issue 5.27.3, Program Status, (identified as Category 2 issues) were on the topic of ageing. CNSC issue 5.27.1 indicated that Bruce Power should, at a minimum, review their ageing management program governing documents against the guidance documents identified in paragraph 4.24 of IAEA NS-G-2.10 [155]; that is, references [165], [166], [167] and [168] included in the 2008 report. CNSC issue 5.27.2 indicated that CNSC staff considered the implementation of a systematic and integrated ageing management program essential for extended plant life. CNSC issue 5.27.3 requested Bruce Power to provide progress on how deficiencies were being addressed. As indicated throughout the 2008 review report, Bruce Power submitted a review against IAEA references and identified the primary steps for an Ageing Management Program [157] to address CNSC issues 5.27.1, 5.27.2, and 5.27.3.

In addition to the ageing-specific CNSC issues 5.27.1, 5.27.2, and 5.27.3, CNSC staff indicated their expectation that the licensee be able to demonstrate meeting the intent of licence condition 3.5 regarding maintenance (identified as a Category 2 issue). Bruce Power committed to completing a maintenance assessment report that will include an assessment of the Bruce Power maintenance program versus the intent of S-210 to address this issue [31]. The maintenance assessment report was submitted to the CNSC in October 2008 [169].

7.3.2. Regulatory Evaluations and Reviews Performed Since 2008

B-REP-00701-27MAY2013-051 [170] provides the results of an assessment of the status of the relevant CNSC inspections that were conducted in the five years since the 2008 review. These inspections are identified in Table 3 of B-REP-00701-27MAY2013-051 [170]. This assessment concluded *inter alia* that Bruce Power should review the status of the recommendations in the following CNSC compliance inspections:

- ID BB 2008 13494 038: Structures, Systems, and Components Monitoring;
- IDB 2008 B 033 TI3082: Bruce B Maintenance Work Execution;
- BRPD A 2012 017: Instrument Air Unit 4 System Inspection; 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.
- BRPD-A-2012-017: Instrument Air Unit 4 System Inspection – Compensatory actions were identified and submitted to the CNSC in NK21-CORR-00531-10012 [171].

In addition to the regulatory evaluations summarized in B-REP-00701-27MAY2013-051 [170], 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 [172]. In their compliance inspection report, BRPD-AB-2014-002, 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. Six action notices and three recommendations were raised as a result of this inspection, as follows:

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-AN03:

"In order to be compliant with NK21-DM-71310 Appendix A Item #2b and flow diagram NK21-DG-71310-00005, Bruce Power is requested to ensure that the information contained in the Bruce A Safety Report, NK21-SR-01320-00002, Table 11-1 regarding the LPSW requirements for the turbine lube-oil coolers is correct"

Action Notice - BRPD-AB-201 4-002-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-201 4-002-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 prior to 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-201 4-002-R01:

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

Recommendation - BRPD-AB-201 4-002-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 six Action Notices and three Recommendations arising from this inspection are provided in [173]. CNSC staff are reviewing Bruce Power's responses.

CNSC staff also conducted a Type II compliance inspection from April to September 2014 on Bruce Power's management of the Bruce A Unit 3 planned maintenance outage [107]. This inspection confirmed that the Bruce Power outage maintenance backlog reduction met requirements, but found that improvements are needed to drive down the backlogs. No formal actions were created as a result of this inspection.

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.

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 objective of the Bruce A ISR is to conduct a safety review of Bruce A and provide input to a practicable set of improvements to be conducted during the Major Component Replacement in Units 3 and 4, 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.

Table 12 summarizes the key issues arising from the Integrated Safety Review of Safety Factor 4.

Table 12: Key Issues

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

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 A.

9. References

[1] 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.

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

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

[4] CNSC RD-360, Life Extension of Nuclear Power Plants, 2008.

[5] NK21-CORR-00531-03839, Bruce A Environmental Assessment Study Report, December 7, 2005.

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

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

[9] 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.

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

[11] 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.

[12] 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.

[13] 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.

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

[15] NK21-CORR-00531-11272, Nuclear Power Reactor Operating Licence, Bruce Nuclear Generating Station A (PROL 15.00/2015), May 1, 2014.

[16] NK21-CORR-00531-11391, Bruce Nuclear Generating Station A Nuclear Power Reactor Operating Licence: Licence Conditions Handbook (LCH-BNGSA-R8), June 4, 2014.

[17] PROL 18.00/2020, e-Doc 4723908, Nuclear Power Reactor Operating Licence, Bruce Nuclear Generating Station A and Bruce Nuclear Generating Station B, June 1, 2015.

[18] LCH-BNGS-R000, e-Doc 4659316, Bruce Nuclear Generating Station A and Bruce Nuclear Generating Station B Power Reactor Operating Licence, June 1, 2015.

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

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

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

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

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

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

[25] 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).

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

[27] CNSC S-98-R1, Reliability Programs for Nuclear Power Plants, July 2005.

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

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

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

[31] 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.

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

[33] NK21-CORR-00531-10456/NK29-CORR-00531-10857/e-Doc 4094940, Application Requirements for Renewal of Power Reactor Operating Licences for Bruce Nuclear Generating Stations A and B, CNSC Letter, R. Lojk to F. Saunders, April 25, 2013.

[34] CAN/CSA-N285.5-08, Periodic Inspection of CANDU Nuclear Power Plant Containment Components, 2008.

[35] 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.

[36] 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.

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

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

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

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

[41] 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.

[42] 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.

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

[44] CAN/CSA-N291-08, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants, Update No. 2 June 2011, Reaffirmed 2013.

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

[46] BP-PROG-11.01-R004, Equipment Reliability, Bruce Power, October 8, 2013.

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

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

[49] BP-PROC-00778-R001, Scoping and Identification of Critical SSCs, Bruce Power, April 11, 2013.

[50] BP-PROC-00779-R000, Continuing Equipment Reliability Improvement, Bruce Power, December 6, 2011.

[51] BP-PROC-00780-R001, Preventive Maintenance Implementation, Bruce Power, August 20, 2014.

[52] BP-PROC-00781-R002, Performance Monitoring, Bruce Power, September 22, 2012.

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

[54] BP-PROC-00584-R006, PASSPORT Equipment Data Management, Bruce Power, December 4, 2012.

[55] BP-PROC-00666-R003, Component Categorization, Bruce Power, September 12, 2014.

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

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

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

[59] BP-PROC-00534-R002, Technical Basis Assessment, Bruce Power, May 27, 2013.

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

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

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

[63] BP-PROC-00603-R002, Preventive Maintenance Just-in-Time [JIT] Review Process, Bruce Power, June 26, 2014.

[64] BP-PROC-00284-R003, Predictive Maintenance, Bruce Power, July 5, 2013.

[65] DPT-PE-00008-R005, System/Component Performance Monitoring Plan, Bruce Power, August 21, 2013.

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

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

[68] BP-PROC-00496-R004, Trouble Shooting Plant Equipment, Bruce Power, June 27, 2014.

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

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

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

[72] BP-PROG-12.02-R005, Chemistry Management, Bruce Power, March 26, 2012.

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

[74] BP-PROC-00060-R026, Station Condition Record Process, Bruce Power, July 22, 2014.

[75] BP-PROC-00019-R009, Action Tracking, Bruce Power, October 4, 2012.

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

[77] DPT-CHM-00003-R006, Control of Chemistry, Bruce Power, May 27, 2013.

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

[79] DPT-CHM-00008-R005, Outage Chemistry Program, Bruce Power, August 28, 2013.

[80] BP-PROC-00017-R002, Major Component Outage Scope Management, Bruce Power, December 19, 2013.

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

[82] BP-PROC-00361-R001, In-Service Testing and Inspection to Satisfy CAN/CSA-N287.7-08 Requirements, Bruce Power, September 15, 2011.

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

[84] BP-PROC-00936-R000, Asset Management Planning, Bruce Power, December 2013.

[85] BP-PROC-00899-R000, Asset Life Projections and Options, Bruce Power, May 2013.

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

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

[88] BP-PROC-00825-R000, Buried Piping Inspection Program, Bruce Power, August 21, 2013.

[89] NK21-PIP-03641.2-00001, Bruce A Periodic Inspection Plan for Unit 1, March 2010.

[90] NK21-PIP-03641.2-00002, Bruce A Periodic Inspection Plan for Unit 2, March 2010.

[91] NK21-PIP-03641.2-00003, Bruce A Periodic Inspection Plan for Unit 3, October 2011.

[92] NK21-PIP-03641.2-00004, Bruce A Periodic Inspection Plan for Unit 4, October 2011.

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

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

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

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

[97] NK21-PIP-03642-00001-R001, Bruce NGS A N285.5 Periodic Inspection Plan for Unit 0A and Units 1 to 4 Containment Components, July 2013.

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

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

[100] BP-PROC-00768-R000, Predictive Maintenance Infrared Thermography Program, Bruce Power, August 30, 2011.

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

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

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

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

[105] BP-PROC-00329-R015, On-Line Work Management Process, Bruce Power, September 23, 2014.

[106] 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

[107] NK21-CORR-00531-11660, CNSC Type II Compliance Inspection Report: BRPD-A-2014-003 Bruce A Unit 3 Maintenance Outage Inspection, CNSC Letter, K. Lafrenière to F. Saunders, October 20, 2014.

[108] NK21-CORR-00531-11698, 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.

[109] NK21-CORR-00531-10818, 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.

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

[111] DPT-PE-00009-R001, System and Component Performance Monitoring Walkdowns, December 18, 2014.

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

[113] 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.

[114] 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.

[115] 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.

[116] 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.

[117] 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.

[118] BP-PROG-03.01-R015, Document Management, Bruce Power, August 18, 2014.

[119] BP-PROC-00068-R022, Controlled Document Life Cycle Management, Bruce Power, December 12, 2014.

[120] BP-PROC-00098-R014, Records Management, Bruce Power, December 5, 2013.

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

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

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

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

[125] 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.

[126] 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.

[127] FORM-10700-R021, Design Scoping Checklist.

[128] BP-PROC-00539-R015, Design Change Package, Bruce Power, August 26, 2014.

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

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

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

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

[133] NK21-CORR-00531-11657, Bruce A Unit 3: Reports on 2014 Pressure Tube Hydrogen Isotope Concentration Measurements, Bruce Power Letter, F. Saunders to K. Lafrenière, November 11, 2014.

[134] NK21-CORR-00531-11339/NK29-CORR-00531-11742, 2014 Annual COG Research and Development Reporting, Bruce Power Letter, F. Saunders to K. Lafrenière, June 16, 2014,

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

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

[137] DPT-ENG-00019-R004, Disposition of PIP/In-Service Inspection Results

[138] 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.

[139] BP-OPP-00002-R012, Operating Policies and Principles – Bruce A, Bruce Power, July 24, 2013.

[140] BP-PROG-05.01-R013, Supply Chain, Bruce Power, March 6, 2014.

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

[142] SEC-PE-00015-R004, Selection of Item Shelf Life Requirements, July 31, 2008.

[143] DPT-MM-00007-R010, Item Shelf Life Management, July 20, 2011.

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

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

[146] NK21-PLAN-34320-00003-R001, Bruce A Vacuum Building FRP Spray Header Life Cycle Management Plan, September 2011.

[147] NK21-PIP-25100-00001-R002, CSA N287.7-08 Periodic Inspection Program for Bruce NGS A Vacuum Building, September 2014.

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

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

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

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

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

[153] CAN/CSA-N286-05, Management System Requirements for Nuclear Facilities.

[154] AU-2006-00015, Periodic Inspection Program Audit, Bruce Power Audit Report, April 2006.

[155] IAEA NS-G-2.10, Periodic Safety Review of Nuclear Power Plants, IAEA Safety Standards Series, Safety Guide, August 2003.

[156] NK21-CORR-00531-04994, Bruce Units 1&2 Return to Service and Life Extension – CNSC Staff Review of Bruce Power's Systematic Review of Safety, CNSC letter, D.A. Desjardins to F. Saunders, May 18, 2007.

[157] NK21-CORR-00531-05738 (Enclosure 1), Bruce A Units 1 and 2 Return to Service – Category 2 Issue Related to Ageing Management, Bruce Power Letter, F. Saunders to P. Elder, February 29, 2008.

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

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

[160] CMD 08-M37/eDoc 3234690, Annual CNSC Staff Report for 2007 on the Safety Performance of the Canadian Nuclear Power Industry, June 10, 2008.

[161] NK21-CORR-00531-06119, AI 080707: Bruce A and B Maintenance Backlog, Bruce Power Letter, F. Saunders to P. Elder, June 9, 2008.

[162] PROL 15.10/2009, NK21-CORR-00531-05996, Nuclear Power Reactor Operating Licence, Bruce Nuclear Generating Station A, March 20, 2008.

[163] NK21-CORR-00531-03477, Bruce Power Periodic Inspection Program Review, Bruce Power Letter, F. Saunders to P. Webster, June 16, 2005.

[164] NK21-PIP-21100-00001-R003, CSA N287.7-08 Periodic Inspection Program for Bruce NGS A Concrete Containment Structures and Appurtenances (Excluding Vacuum Building), October 2014.

[165] IAEA Technical Report Series No. 338, Methodology for the Management of Ageing of Nuclear Power Plant Components Important to Safety, July 1992.

[166] IAEA Safety Practices Safety Series No. 50-P-3, Data Collection and Record Keeping for the Management of Nuclear Power Plant Ageing, December 1991.

[167] IAEA Safety Report Series No. 15, Implementation and Review of Nuclear Power Plant Ageing Management Programme, April 1999.

[168] IAEA TECDOC-1197, Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: CANDU Reactor Assemblies, February 2001.

[169] NK21-CORR-00531-05991, Bruce A Units 1 and 2 Return to Service: Integrated Safety Review – Category 2 Issue 7, Bruce Power Letter, F. Saunders to K. Lafrenière, October 31, 2008.

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

[171] NK21-CORR-00531-10012, Responses to CNSC Comments on the Bruce A 2011 Annual Reliability Report, Letter from F. Saunders to R. Lojk, November 22, 2012.

[172] 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.

[173] 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.

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

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

N285.4-05 [32], Periodic Inspection of CANDU Nuclear Power Plant Components is listed as condition 4.3(i) (a) in the PROL [15]. A newer version of this standard was issued in 2009 with Update 1 in 2010 and Update 2 in 2011, and CNSC has indicated that the 2009 version with Update No. 2 issued in June 2011 will be included in the next PROL [33]. 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 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 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 12.

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, N285.5-08, Periodic Inspection of CANDU Nuclear Power Plant Containment Components is listed as condition 4.3(i) (b) in the PROL [15], and will be retained in the next PROL. 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 [34] and N285.5-13 [23] 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. It should be noted 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 A, 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 (FRP) 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.

Appendix B – Review Against Codes and Standards

This appendix presents the clause-by-clause assessments that are performed for this Safety Factor. The ISR 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 ISR 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 ISR Basis Document defines RNA as "the particular clause provides requirements that are less strenuous than clauses of another standard that has already been assessed". The definition has been broadened to include 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. 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. This procedure defines the elements, functional requirements, implementing procedures and key responsibilities associated with the Nuclear Safety Assessment (NSA) process. The objective of NSA is to ensure that all necessary nuclear safety requirements are defined for the actual or proposed design of the plant throughout the design modification process or in addressing emergent issues (e.g., plant aging) that may affect the Design Basis or the Safety Report Basis. NSA is the systematic process carried out, throughout the design modification process or in addressing emergent issues (e.g., plant aging) that may affect the Design Basis or the	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.	Safety Report Basis, to ensure that all necessary nuclear safety requirements are defined for the actual or proposed design of the plant.	
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	The introductory paragraph in this clause is new and includes reference to aging management. The design documentation follows well established	C

Article No.	Clause Requirement	Assessment	Compliance Category
	eventual decommissioning of the NPP. 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	processes and procedures as described in Design Documentation, BP-PROC-00335. This procedure specifies the design activities and outputs that define and manage the Plant Design Basis such that the nuclear operating stations can operate safely and reliably for the duration of their design life. 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
	criteria 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	Additional information may be found in: • CNSC, RD/GD-369, Licence Application Guide: Licence to Construct a Nuclear Power Plant, 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	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
	determined based on their importance to safety, as determined using the criteria in section 7.1. The design rules should include, as applicable: • identified codes and standards • conservative safety margins • reliability and availability: • material selection • single-failure criterion • redundancy • separation • diversity • independence • fail-safe design • equipment qualification: • environmental qualification • seismic qualification • qualification against electromagnetic interference • operational considerations: • testability		

Article No.	Clause Requirement	Assessment	Compliance Category
	• inspectability • maintainability • aging management • management system The design of complementary design features should be such that they are effective for fulfilling the actions credited in the safety analysis, with a reasonable degree of confidence. Other SSCs that are credited for DECs should also meet this expectation. Design rules should include relevant national and international codes and standards. In cases of SSCs for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar SSCs may be applied; in the absence of such codes and standards, the results of experience, tests, analysis or a combination of these may be applied, and this approach should be justified. A set of design limits consistent with the key physical parameters for each SSC important to safety for the nuclear power plant should be specified for all operational states, DBAs and DECs.		

Article No.	Clause Requirement	Assessment	Compliance Category
	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: 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	A new requirement has been added regarding consideration of aging effects due to service life for SSCs important to safety. The EQ process described in BP-PROC-00261 supports the Design Management procedure BP-PROC-00335 and 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.	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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 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,		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 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.		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 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:		

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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 • 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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. Examples of protective barriers include: • steam-protected rooms and enclosures • steam doors • water-protected rooms (for flooding)		

Article No.	Clause Requirement	Assessment	Compliance Category
	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, Palto 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 (IEC), 60780 ed 2.0, Nuclear Power Plants - Electrical Equipment of the Safety System – Qualification, Geneva, 1998. • IEEE, Standard 323, IEEE Standard for		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 identification and evaluation of potential liquefiable	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
	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)		

Article No.	Clause Requirement	Assessment	Compliance Category
	• DEC loads (or beyond-design loads) The design should identify all DEC loads considered in the structure design and provide the assessment methodology and acceptance criteria. The structural design should withstand, accommodate or avoid foundation settlement (total and differential), according to its performance requirements. The structural design should consider the impact of aging on the structure and its material. The design should include sufficient safety margins for the buildings and structures that are important to safety. The physical and material description of each civil structure and its base slab should include: • the type of structure, and its structural and functional characteristics • the geometry of the structures, including sketches showing plan views at various elevations and sections (at least two orthogonal directions) • the relationship between adjacent structures, including any separation or structural ties		

Article No.	Clause Requirement	Assessment	Compliance Category
	• the type of base slab and its arrangement with the methods of transferring horizontal shears (such as those seismically induced) to the foundation media Containment structure The design should specify the safety requirements for the containment building or system, including, for example, its structural strength, leak tightness, and resistance to steady-state and transient loads (such as those arising from pressure, temperature, radiation, and mechanical impact) that could be caused by postulated internal and external hazards. In addition, the design should specify the safety requirements and design features for the containment internal structures, (such as the reactor vault structure, the shielding doors, the airlocks, and the access control and facilities). The design of the containment structure should include: • base slab and sub-base • containment wall and dome design		

Article No.	Clause Requirement	Assessment	Compliance Category
	• containment wall openings and penetrations • pre-stressing system • containment liner and its attachment method The design pressure of the containment building should be determined by increasing by at least 10% the peak pressure that would be generated by the DBA (refer to clause 4.49 of IAEA NS-G-1.10, Design of Reactor Containment Systems for Nuclear Power Plants). Ultimate internal pressure capacity should be provided for the containment building structures including containment penetrations. If the containment building foundation is a common mat slab which is not separated from the other buildings foundation, the impact should be evaluated. Concrete containment structures should be designed and constructed in accordance with the CSA N287 series, as applicable:		

Article No.	Clause Requirement	Assessment	Compliance Category
	• N287.1, General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, for general requirements in documentation of design specification and design reports • N287.2, Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, for material • N287.3, Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants for design • N287.4, Construction, Fabrication and Installation Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants, and N287.5, Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants, for containment construction and inspection • N287.6, Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants, for pressure test before operation Steel containment structures should be designed according to the ASME Boiler and Pressure Vessel Code, Section III, Division 1, Subsection NE, Class MC Components or equivalent standard. Stability of the containment vessel and appurtenances should be evaluated using ASME Code Case N-284-1,		

Article No.	Clause Requirement	Assessment	Compliance Category
	Metal Containment Shell Buckling Design Methods, Section III, Division 1, Class MC. For other requirements on the design of containment structures, refer to section 8.6.2 of this regulatory document. Safety-related structures The safety-related structures other than the containment should be designed and constructed in accordance with CSA N291, Requirements for safety-related structures for CANDU nuclear power plants. The design of other safety-related structures should include: • internal structures of reactor building • service (auxiliary) building • fuel storage building • control building • diesel generator building		

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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: • 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.		

Article No.	Clause Requirement	Assessment	Compliance Category
	• 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.		
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	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 (RSE), with support from Reactor Safety, Corporate & Station Component Engineers and Design	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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 • 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	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. 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	

Article No.	Clause Requirement	Assessment	Compliance Category
		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 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.	Following is guidance relates to plant aging: The demonstration of thermal margin is expected to be presented in a manner that accounts for all possible reactor operational states and conditions, as determined from operating maps including all AOOs. The demonstration should also include long term effects of plant aging and other expected changes to core configuration over the operating life of the plant. 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	C

Article No.	Clause Requirement	Assessment	Compliance Category
	Fuel design limits shall be established to include, as a minimum, limits on fuel power or temperature, limits on fuel burnup, and limits on the leakage of fission products in the reactor cooling system. The design limits shall reflect the importance of preserving the fuel matrix and cladding, as these are first and second barriers to fission product release, respectively. The design shall account for all known degradation mechanisms, with allowance being made for uncertainties in data, calculations, and fuel fabrication. Fuel assemblies shall be designed to permit adequate inspection of their structures and components prior to and following irradiation. In DBAs, the fuel assembly and its component parts shall remain in position with no distortion that would prevent effective post-accident core cooling or interfere with the actions of reactivity control devices or mechanisms. The design shall specify the acceptance criteria necessary to meet these requirements in DBAs. The requirements for reactor and fuel assembly	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.	

Article No.	Clause Requirement	Assessment	Compliance Category
	design shall apply in the event of changes in fuel management strategy, or in operating conditions, over the lifetime of the plant. Fuel design and design limits shall reflect a verified and auditable knowledge base. The fuel shall be qualified for operation, either through experience with the same type of fuel in other reactors, or through a program of experimental testing and analysis, to ensure that fuel assembly requirements are met. Guidance The fuel design and qualification should provide assurance that the reactor core design requirements in section 8.1 are met. Acceptance criteria should be established for fuel damage, fuel rod failure, and fuel coolability. These criteria should be derived from experiments that identify the limitations of the material properties of the fuel and fuel assembly, and related analyses. The fuel design criteria and other design considerations are discussed below.		

Article No.	Clause Requirement	Assessment	Compliance Category
	Fuel damage Fuel damage criteria should be established for all known damage mechanisms in operational states (normal operation and AOOs). The damage criteria should assure that fuel dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. When applicable, the fuel damage criteria should consider high burnup effects based on irradiated material properties data. The criteria should include stress, strain or loading limits, the cumulative number of strain fatigue cycles, fretting wear, oxidation, hydriding (deuteriding in CANDU reactors), build-up of corrosion products, dimensional changes, rod internal gas pressures, worst-case hydraulic loads, and LWR control rod insertability. Fuel rod failure Fuel rod failure applies to operational states, DBAs and DECs. Fuel rod failure criteria should be provided for all known fuel rod failure mechanisms. The design should ensure that fuel does not fail as a result of specific causes during operational states. Fuel rod failures could occur during DBAs and DECs, and are accounted for in the safety analysis.		

Article No.	Clause Requirement	Assessment	Compliance Category
	Assessment methods should be stated for, fuel failure mechanisms, reactor loading and power manoeuvring limitations, and fuel duty which lead to an acceptably low probability of failure. When applicable, the fuel rod failure criteria should consider high burnup effects, based on data of irradiated material properties. The criteria should include: • hydriding • cladding collapse • cladding overheating • fuel pellet overheating • excessive fuel enthalpy • pellet-clad interaction • stress-corrosion cracking • cladding bursting • mechanical fracturing Fuel coolability Fuel coolability applies to DBAs and, to the extent		

Article No.	Clause Requirement	Assessment	Compliance Category
	practicable, DECs. Fuel coolability criteria should be provided for all damage mechanisms in DBAs and DECs. The fuel should be designed to ensure that fuel rod damage will not interfere with effective emergency core cooling. The cladding temperatures should not reach a temperature high enough to allow a significant metal- water reaction to occur, thereby minimizing the potential for fission product release. The criteria should include cladding embrittlement, fuel rod ballooning, structural deformation and, in CANDU, beryllium braze penetration. Other considerations The design should also include: • all expected fuel handling activities • the effects of post-irradiation fuel assembly handling • cooling flow of other components of LWR fuel assembly (such as control rods, poison rods, instrumentation, or neutron sources) Testing, inspection, and surveillance programs		

Article No.	Clause Requirement	Assessment	Compliance Category
	Programs for testing and inspection of new fuel, as well as for online fuel monitoring and post-irradiation surveillance of irradiated fuel should be established. Fuel specification The design should establish the specification of fuel rods and assembly (including LWR control rods) in order to minimize design deviations and to determine whether all design bases are met (such as limits and tolerances). Reactor core thermal hydraulic design The thermalhydraulic design should be such that sufficient margin exists with regard to maintaining adequate heat transfer from the fuel to the reactor coolant system, to prevent fuel sheath overheating. The design requirements can be demonstrated by meeting a set of derived acceptance criteria, as required by REGDOC-2.4.1, Deterministic Safety Analysis.		

Article No.	Clause Requirement	Assessment	Compliance Category
	Critical heat flux (CHF) is defined as the heat flux at departure from nucleate boiling (DNB), commonly used in pressurized water reactors (PWRs), or at dryout, commonly used in CANDU designs. It should be noted that, although a thermal margin criterion is sufficient to demonstrate that overheating from a deficient cooling mechanism can be avoided; other mechanistic methods may be acceptable as CHF is not considered as a failure mechanism. In some designs, CHF conditions during transients can be tolerated if it can be shown by other methods that the sheath temperatures do not exceed well-defined acceptable limits. However, any other criteria than the CHF criterion should address sheath temperature, pressure, time duration, oxidation, embrittlement etc., and these new criteria should be supported by sufficient experimental and analytical evidence. In the absence of such evidence, the core thermal-hydraulic design is expected to demonstrate a thermal margin to CHF. The demonstration of thermal margin is expected to be presented in a manner that accounts for all possible reactor operational states and conditions, as determined from operating maps including all AOOs. The demonstration should also include long term effects of plant aging and other expected changes to core configuration over the operating life		

Article No.	Clause Requirement	Assessment	Compliance Category
	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: • ANSI/ANS, 57.5, Light Water Reactor Fuel Assembly Mechanical Design and Evaluation, La		

Article No.	Clause Requirement	Assessment	Compliance Category
	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. Operating conditions in which components of the	There is a design requirement to take into account all conditions of the boundary material in normal operation (including maintenance and testing), AOOs, 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. 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	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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 The design should have adequate provisions with	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 SSCs are identified in Life Cycle Management Plans. In particular, ageing mechanisms for 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
	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		

Article No.	Clause Requirement	Assessment	Compliance Category
	adequacy of the following: • 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		

Article No.	Clause Requirement	Assessment	Compliance Category
	systems: 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:		

Article No.	Clause Requirement	Assessment	Compliance Category
	• flow rate in single and two phase flow • consideration of corrosion of valve surfaces • provisions for ensuring that relief discharge 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)		

Article No.	Clause Requirement	Assessment	Compliance Category
	• vibration monitoring provisions Additional information 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	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	C

Article No.	Clause Requirement	Assessment	Compliance Category
	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 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,	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 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
	Deterministic Safety Analysis. Trip coverage should be demonstrated across the full range of operating states, for all credited shutdown means and all credited trip parameters. Note that the number of credited shutdown means and the number of credited trip parameters can vary with the event, the reactor design, and whether there is a direct trip available. Defining derived acceptance criteria appropriate to a particular design is the responsibility of the design authority. CNSC REGDOC-2.4.1, Deterministic Safety Analysis, provides the requirements. Derived acceptance criteria should be defined separately for AOOs and DBAs. The derived acceptance criteria should be set to give an appropriate level of confidence that a fundamental safety function is assured, or that a barrier to fission product release will not fail. The derived acceptance criteria should: • be quantifiable and well understood • account for the fact that the safety analysis is stylized, and the plant condition at the time of the		

Article No.	Clause Requirement	Assessment	Compliance Category
	accident may be significantly different from the analyzed state • cover uncertainties in analysis, input plant and analysis parameters, as well as code validation Direct trips are the preferred means of actuating a shutdown means, due to their robustness and low dependence on calculational models. Diverse trip parameters measure different physical variables on the reactor, thus providing additional protection against common mode failure. Where it is impracticable to provide full diversity of trip parameters, different measurement locations, different instrument types and different processing computers should be provided. Manual trip is considered an acceptable trip parameter, if the operator has adequate time to initiate the shutdown action following unambiguous indication of the need to perform the action (in accordance with section 8.10.4). It is the responsibility of the design authority to identify and justify those trip parameters that can be considered “direct”. The design authority should also demonstrate that any trip parameters that are a measure of the event, but not a measure of the challenge to acceptance criteria, cannot be		

Article No.	Clause Requirement	Assessment	Compliance Category
	"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 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	This clause includes new requirements to account for postulated aging effects and demonstrate sufficient design margins.	C

Article No.	Clause Requirement	Assessment	Compliance Category
	report. The preliminary safety analysis shall assist in the establishment of the design-basis requirements for the items important to safety, and demonstrate whether the plant design meets applicable requirements. The final safety analysis shall: 1. reflect the as-built plant 2. account for postulated aging effects on SSCs important to safety 3. demonstrate that the design can withstand and effectively respond to identified PIEs 4. demonstrate the effectiveness of the safety systems and safety support systems 5. derive the OLCs for the plant, including:	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 RIH temperature, PT diametral creep, etc. over 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
	a. operational limits and set points important to safety b. allowable operating configurations, and constraints for operational procedures 6. establish requirements for emergency response and accident management 7. determine post-accident environmental conditions, including radiation fields and worker doses, to confirm that operators are able to carry out the actions credited in the analysis 8. demonstrate that the design incorporates sufficient safety margins 9. confirm that the dose and derived acceptance criteria are met for all AOOs and DBAs 10. demonstrate that all safety goals have been met Guidance		

Article No.	Clause Requirement	Assessment	Compliance Category
	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 previous version assessed (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 Bruce A PROL specifies that the licensee shall implement and maintain a plant design basis management system such that the structures, systems and components continue to meet the design basis and the plant can operate safely for the full duration of its design life. Design basis management 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 addressed, care and attention to good fabrication and construction practices are inherent to minimize the impact of construction on the component.	C

Article No.	Clause Requirement	Assessment	Compliance Category
		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 Bruce A 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 acid rain.	

Article No.	Clause Requirement	Assessment	Compliance Category
		- 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 main containment structure is 1.0% contained mass/hr at the design pressure of the structure. However, the OP&P value is 2.0%/hr at the design pressure of the structure, and the Bruce A safety analysis value is 2.32 %/h + 12 cm^2 turbulent component. 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 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	C

Article No.	Clause Requirement	Assessment	Compliance Category
		containment boundary 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 PIP results obtained in accordance with N285.5 and N285.7, leakage rate test results, Plant Health reports, SCRs, OPEX, etc. It should be noted that inspection and testing of containment structures are part of the Bruce A PROL, and specified limits are	C

Article No.	Clause Requirement	Assessment	Compliance Category
		included in the Operating Policies and 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 N291-08 (R2013), Requirements for Safety-Related Structures for CANDU 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 further assessment is performed in “Safety Factor 1– Plant Design”.

Table B3: CSA N291-08, Requirements for Safety-Related Structures for CANDU Nuclear Power Plants

Article No.	Clause Requirement	Assessment	Compliance Category
7.3.2.1	The engineer shall establish an in-service examination program to protect the structure during the life of the plant. 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.	Bruce Power is developing an in-service examination program document for safety related structures. This program is expected to be in place by March 2015. As an interim transition measure for compliance with N291-08, Bruce Power has developed "CSA N291 In-Service Inspection Program for Bruce NGS A Safety Related Structures", NK21-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) locations of components to be examined; (d) methods of examination and testing; (e) frequency and amount (i.e., statistical distribution) of examination and testing;	The scope of "CSA N291 In-Service Inspection Program for Bruce NGS A Safety Related Structures", NK21-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 and reporting of results 2) General requirements for personnel qualifications, basis of comparisons and	C

Article No.	Clause Requirement	Assessment	Compliance Category
	(f) acceptance criteria; and (g) reporting and documentation requirements. Note: Special consideration should be given to factors such as accessibility, effectiveness, and accuracy of examination methods.	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.	
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. Note: 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 A Safety Related Structures", NK21-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 Primary Irradiated Fuel Bay Service Building Ancillary Service Building Secondary Irradiated Fuel Bay	C

Article No.	Clause Requirement	Assessment	Compliance Category
		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 NK21-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 - leaking or broken window and door seals - condition of doors, windows and framing	

Article No.	Clause Requirement	Assessment	Compliance Category
		- 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/licensee and the regulatory authority. 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 4.6 of NK21-PIP-20000-00001, "CSA N291 In-Service Inspection Program for Bruce NGS A 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 NK21-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 NK21-PIP-20000-00001, "CSA N291 In-Service Inspection Program for Bruce NGS A 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
7.3.4	Following any abnormal/environmental condition, all structural components shall be subjected to a visual	NK21-PIP-20000-00001, "CSA N291 In-Service Inspection Program for Bruce NGS A Safety	Gap

Article No.	Clause Requirement	Assessment	Compliance Category
	inspection and other methods of examination, as required, to evaluate the integrity of the structure.	Related Structures" does not describe inspection requirements following an abnormal/environmental condition.	

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 previous version assessed (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