FORM-14159 R000*

Periodic Safety Review - Final Document Review Traveler

Bruce Power Document #:	Revision:	Information Classification	Usage Classification
NK29-SFR-09701-00006	R000	Internal Use Only	Information

Bruce Power Document Title:
Safety Factor 6 – Probabilistic Safety Analysis

Bruce Power Contract/Purchase Order:	Bruce Power Project #:
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Supplier's Name:	Supplier Document #:	Revision:
CANDESCO	K-421231-00206	R00

Supplier Document Title:
Safety Factor 6 – Probabilistic Safety Analysis

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

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

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

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

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Division of Kinectrics Inc.

NK29-SFR-09701-00006

Title: Safety Factor 6 - Probabilistic Safety Analysis

File: K-421231-00206-R00

A Report Submitted to Bruce Power

September 20, 2016

Issue R00D0	**Reason for Issue:** For harmonization				
	Author: R. Ion	Verifier:	Reviewer:	Approver:	Date:
Issue R00D1	**Reason for Issue:** For internal review				
	Author: R. Ion	Verifier:	Reviewer: L. Watt G. Archinoff	Approver:	Date: Mar 23, 2016
Issue R00D2	**Reason for Issue:** For Bruce Power review				
	Author: R. Ion	Verifier: G. Buckley	Reviewer: L. Watt G. Archinoff	Approver:	Date: May 13, 2016
Issue R00D3	**Reason for Issue:** Addresses Bruce Power review comments and internal verification comments.				
	Author: R. Ion	Verifier: G. Buckley	Reviewer: L. Watt G. Archinoff	Approver:	Date: August 8, 2016

Issue	**Reason for Issue:**				
R00	For use				
	Author: R. Ion	Verifier: G. Buckley	Reviewer: L. Watt G. Archinoff	Approver: L. Watt	Date: Sept 20, 2016

Document Classification:	Security Classification:
Report	Client Proprietary

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Acronyms and Abbreviations

APU	Actual Past Unavailability
BBRA	Bruce B Risk Assessment
BP	Bruce Power
CA	Computational Aids
CAFTA	Computer Aided Fault Tree Analysis
CANDU	Canada Deuterium Uranium
CCF	Common Cause Failure
CFAM	Corporate Functional Area Manager
CFF	Containment Failure Frequency
CNSC	Canadian Nuclear Safety Commission
COG	CANDU Owners Group
CSA	Canadian Standards Association
EME	Emergency Mitigating Equipment
ET	Event Tree
FASA	Focus Area Self-Assessment
FDC	Fuel Damage Category
FT	Fault Tree
FV	Fussell-Vesely
GAR	Global Assessment Report
HI	Human Interaction
IAEA	International Atomic Energy Agency
IEs	Initiating Events
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
LRF	Large Release Frequency
LTEP	Long Term Energy Plan
MAAP	Modular Accident Analysis Program
MCR	Major Component Replacement
MSM	Management System Manual

NPP	Nuclear Power Plant
NSCA	Nuclear Safety and Control Act
PDSs	Plant Damage States
PFU	Predicted Future Unavailability
PIE	Postulated Initiating Event
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment (synonymous with PRA)
PSR	Periodic Safety Review
QA	Quality Assurance
RAW	Risk Achievement Worth
RCs	Release Categories
SAM	Severe Accident Management
SAMG	Severe Accident Management Guidelines
SCA	Safety Control Area
SCDF	Severe Core Damage Frequency
SFR	Safety Factor Report
SIS	Systems Important to Safety
SRF	Small Release Frequency
SSC	Structure, System and Component

1. Objective and Description

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

(i) Prepare and conduct a periodic safety review;

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

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

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

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

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

1.1. Objective

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

The specific objectives of the review of this Safety Factor [1] are to determine:

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

1.2. Description

The review is conducted in accordance with the Bruce B PSR Basis Document [5], which states that the review tasks are as follows:

1. The existing PSA, including the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case;
2. Whether accident management programs for accident conditions (design basis accident conditions and design extension conditions) are consistent with PSA models and results;
3. Whether the scope and applications of the PSA are sufficient;
4. The status and validation of analytical methods and computer codes used in the PSA;
5. Whether the results of PSA show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria; and
6. Whether the existing scope and application of the PSA are sufficient for its use to assist the Integrated Safety Review (ISR) global assessment, for example, to compare proposed improvement options.

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

[1] This Safety Factor is entitled “Probabilistic Safety Analysis”. However, Probabilistic Safety Analysis is referred to as Probabilistic Safety Assessment (PSA) throughout the document; moreover, Probabilistic Risk Assessment (PRA) is equivalent to PSA.

2. Methodology of Review

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

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

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

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

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

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

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

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

micro-gaps are consolidated into comprehensive statements that describe the deviation known as Safety Factor macro-gaps, which are listed in Section 8 of the Safety Factor Reports, as applicable.

3. Applicable Codes and Standards

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

3.1. Acts and Regulations

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

3.2. Power Reactor Operating Licence

The list of codes and standards related to probabilistic safety analysis that are referenced in the PROL [1] and LCH [2], and noted in Table C-1 of the Bruce B PSR Basis Document [5], are identified in Table 1. The edition dates referenced in the third column of the table are the modern versions used for comparison. The PROL contains the following clauses pertinent to this Safety Factor:

- Licence Condition 4.1 of the Operating Licence [1] states that the licensee shall implement and maintain a probabilistic safety assessment program. Per the LCH [2], CNSC REGDOC-2.4.2 [23] outlines the requirements related to PSA.
- Licence Condition 4.2 of the Operating Licence [1] states that the licensee shall ensure that design and analysis computer codes and software used to support the safe operation of the nuclear facilities are of adequate quality. Per the LCH [2], CSA N286.7-99 [24] provides the specific requirements related to the development, modification, maintenance and use of computer programs used in analytical, scientific and design applications.
- Licence Condition 5.1 of the Operating Licence [1] states that the licensee shall implement and maintain a design program. As part of the design basis management, plant status changes are controlled such that the plant is maintained and modified within the limits prescribed by the design and licensing basis. This includes safety and control measures for changes that could introduce hazards different in nature or greater in

probability or consequence than those considered by the probabilistic safety assessment.

- Licence Condition 6.1 of the Operating Licence [1] states that the licensee shall implement and maintain programs to ensure fitness for service of systems, structures and components. Per the LCH [2], this includes a proper reliability program and implementation to ensure that Systems Important to Safety continue to meet their performance requirements. CNSC RD/GD-98 [25] outlines the requirements for a reliability program.

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

Document Number	Document Title	Modern Version Used for PSR Comparison	Type of Review
CNSC REGDOC-2.3.3	Periodic Safety Reviews	[3]	NA
CNSC REGDOC-2.4.2 (2014)	Probabilistic Safety Assessment for Nuclear Power Plants	[23]	CBC
CNSC RD/GD-98 (2012)	Reliability Programs for Nuclear Power Plants	[25]	NA
CSA N286.7-99	Quality Assurance of Analytical, Scientific And Design Computer Programs for Nuclear Power Plants	[24]	NA
CSA N286-05 [26]	Management System Requirements for Nuclear Facilities	CSA N286-12 [27]	NA
CSA N290.15-10	Requirements for the Safe Operating Envelope of Nuclear Power Plants	[28]	NA
Assessment type: **NA**: Not Assessed; **CBC**: Clause-by-Clause; **PCBC**: Partial Clause-by-Clause; **CTC**: Code-to-Code; **HL**: High Level; **2SF**: Assessment performed in another SFR; **CV**: Confirm Validity of Previous Assessments			

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

CNSC REGDOC-2.4.2: CNSC REGDOC-2.4.2, Probabilistic Safety Assessment for Nuclear Power Plants [23] sets out the CNSC requirements with respect to Probabilistic Safety

Assessment. It supersedes CNSC document S-294 [29]. CNSC REGDOC-2.4.2 includes amendments to reflect lessons learned from the Fukushima nuclear event of March 2011, as applicable to PSA. CNSC REGDOC-2.4.2 [23] sets out 10 high-level requirements on the scope, quality and frequency of Probabilistic Risk Assessment (PRA) activities to be conducted by the licensee of an NPP in Canada. In comparison with S-294, CNSC REGDOC-2.4.2 contains additional guidance clauses that elaborate further on the requirements or provide direction on how to meet the requirements. Bruce Power is transitioning to REGDOC-2.4.2 for PSA over the current licence period and has a plan in place (see the LCH [2]) to meet the full compliance with it by the June 30, 2019, target date. In view of the importance of CNSC REGDOC-2.4.2 as the primary regulatory document for PSA, a clause-by-clause review was conducted against this standard and the results are included in Appendix B (B.1).

CNSC RD/GD-98: Table C-1 of the PSR Basis Document [5] specifies that CNSC RD/GD-98 [25], Reliability Programs for Nuclear Power Plants, does not need to be assessed. The LCH [2] (page 54) states that Bruce Power has prepared an implementation plan to transition to the requirement of RD/GD-98 that includes the mapping between the existing PRA RD/GD-98 requirements and the Equipment Reliability program document. Per the LCH [2], Bruce Power was targeting completion for December 2015. The latest version of the Equipment Reliability program document [30] includes in its Appendix C the program's procedures mapping to RD/GD-98.

CSA N286-12: CSA N286-05 is noted in the PROL (Licence Condition 1.1 [1]). Per the LCH [2], an implementation strategy for the 2012 version is in progress to be submitted to the CNSC by the end of January 2016. CNSC staff have stated that in their view the CSA N286-12 version of CSA N286 "does not represent a fundamental change to the current Bruce Power Management System" and have acknowledged that "the new requirements in CSA N286-12 are already addressed in Bruce Power's program and procedure documentation" [31].

Bruce Power had agreed to perform a gap analysis and to prepare a detailed transition plan, and to subsequently implement the necessary changes in moving from the CSA N286-05 version of the code to the CSA N286-12 version, during the current licensing period [32]. 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 [33]. Bruce Power further stated CSA N286-12 does not establish any significant or immediate new safety requirements that would merit a more accelerated implementation. The gap analysis and the resulting transition plan were submitted to the CNSC [34]. Per [34], the major milestones of the transition plan to N286-12 are as follows:

- 22 January 2016: Discuss all the regulatory actions and the transition plan at the Corporate Functional Area Manager (CFAM) meeting
- 31 December 2016: Revision of CFAM Program Document(s) [with LCH notification requirements to the CNSC] to comply with CSA N286-12 requirements completed.
- 31 March 2017: Revision of CFAM Program Document(s) [that do not have LCH notification requirements to the CNSC] to comply with CSA N286-12 requirements completed

- 31 December 2017: Confirmation that that all impacted documents in the program suite comply with the requirements of CSA N286-12
- 15 September 2018: Verification via a Focused Area Self-Assessment (FASA) that previously identified transition Gaps to meeting the requirements of CSA N286-12 have been addressed and effectively implemented
- 14 December 2018: issue notification to the CNSC regarding state of CSA N286-12 readiness, and, implementation date

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

CSA N286.7-99: CSA N286.7-99 [24] has been assessed as part of the 2013 interim PSR and has not changed since this assessment. A CNSC letter [35] also acknowledges that the standard CSA N286.7-99 or equivalent Quality Assurance (QA) computer code requirements are being followed by Bruce Power. Furthermore, CSA N286.7-99 is in the current licence [2] and accordingly no further assessment against its requirements is performed in this Safety Factor report.

CSA N290.15-10: CSA N290.15 [28] is the first edition of CSA standard for requirements for the safe operating envelope of nuclear power plants. This Standard provides requirements for the definition, implementation, and maintenance of the safe operating envelope at nuclear power plants. In addition, guidance material for existing Canada Deuterium Uranium (CANDU) nuclear power plants is provided in Annex A to support the requirements. Per Licence Condition 3.1 of [2], "Bruce Power still has to update several of their program documents and complete the associated training requirements before they are fully compliant with N290.15. The completion date for these administrative updates is September 30, 2015." Most of the program documents committed by Bruce Power [36] to make changes to, were updated by September 30, 2015. The only outstanding program document, BP-PROG-12.03, Nuclear Fuel Management, was to be updated by the end of 2015 [37]. BP-PROG-12.03 [38], has been updated and issued on January 29, 2016, as revision R004. This revision now makes reference to CSA N290.15-10, Requirements for the Safe Operating Envelope of Nuclear Power Plants. This Safety Factor Report does not include any further assessment or discussion of the CSA N290.15-10 standard.

3.3. Regulatory Documents

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

Table 2: Regulatory Documents

Document Number	Document Title	Reference	Type of Review
CNSC REGDOC-2.5.2 (2014)	Design of Reactor Facilities: Nuclear Power Plants	[39]	PCBC
Assessment type: **NA**: Not Assessed; **CBC**: Clause-by-Clause; **PCBC**: Partial Clause-by-Clause; **CTC**: Code-to-Code; **HL**: High Level; **2SF**: Assessment performed in another SFR; **CV**: Confirm Validity of Previous Assessments			

CNSC REGDOC-2.5.2: CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants [39], sets out requirements and guidance for new licence applications for NPPs. It establishes a set of comprehensive design requirements and guidance that are risk-informed and align with accepted international codes and practices. The sections of REGDOC-2.5.2 that are relevant to PSA are assessed in Appendix B (B.2), while a more comprehensive assessment of the document is performed in the report on Safety Factor 1 - Plant Design.

3.4. CSA Standards

Per Table C-1 of the PSR Basis Document [5], there are no other Canadian Standards Association (CSA) standards identified in the Bruce Power PROL [1] and LCH [2] for inclusion in this Safety factor review.

3.5. International Standards

The international standard listed in Table 3 is relevant to this Safety Factor and was considered for this review.

Table 3: International Standards

Document Number	Document Title	Reference	Type of Review
IAEA SSG-25 (2013)	Periodic Safety Review for Nuclear Power Plants	[40]	NA
Assessment type: **NA**: Not Assessed; **CBC**: Clause-by-Clause; **PCBC**: Partial Clause-by-Clause; **CTC**: Code-to-Code; **HL**: High Level; **2SF**: Assessment performed in another SFR; **CV**: Confirm Validity of Previous Assessments			

IAEA SSG-25: IAEA SSG-25 [40] addresses the periodic safety review of nuclear power plants. Per the PSR Basis Document [5], this PSR is being conducted in accordance with REGDOC-2.3.3. As stated in REGDOC-2.3.3 [3], this regulatory document is consistent with

IAEA SSG-25. The combination of IAEA SSG-25 and REGDOC-2.3.3, define the review tasks that should be considered for the Safety Factor Reports. However, no assessment is performed specifically on IAEA SSG-25.

3.6. Other Applicable Codes and Standards

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

4. Overview of Applicable Bruce B Station Programs and Processes

Within the organization of Bruce Power's programs and processes, probabilistic safety analysis falls under the broader function of Nuclear Safety Assessment, which also covers activities such as deterministic safety analysis, hazard analysis and criticality safety assessment. The Nuclear Safety Assessment function, together with the Design Management Function, falls under Bruce Power's Plant Design Basis Management Program.

Nuclear safety is addressed at the highest level of the hierarchy in the Management System Manual (BP-MSM-1-R012) [41]. The high level policies described in the MSM find expression in the program on Plant Design Basis Management [42]. In addition, the boundaries within which the station may be operated safely are outlined for Bruce B in the Operating Policies and Principles BP-OPP-00001 [43]. The program is implemented through the following governance documents:

- BP-PROC-00363 on Nuclear Safety Assessment [44];
- BP-PROC-00335 on Design Management [45];
- BP-PROC-00582 on Engineering Fundamentals [46];
- BP-PROC-00502 on Resolution of Differing Professional Opinions [47];
- DIV-ENG-00008 on Engineering Work Management [48];
- DIV-ENG-00009 on Design Authority [49]
- DIV-ENG-00021 on Professional Engineering Accountabilities [50].

The first document, BP-PROC-00363 [44], is particularly relevant to this review of Safety Factor 6. Regarding PRA, the implementation of BP-PROC-00363 [44] on Nuclear Safety Assessment is supported by DIV-ENG-00010 [51] Probabilistic Risk Assessment Process, the purpose of which is to establish a process for the evaluation of the safe operation of the station utilizing PRA and comparing the results against established safety goal targets and limits.

Four lower level procedures define the actual required processes in PRA applications to ensure safe operation:

- Procedure DPT-RS-00007 [52], Preparation and Maintenance of Probabilistic Risk Assessments, defines the process for preparing and maintaining a PRA. The maintenance aspect is to support the continuing use of PRA in the conduct of engineering, maintenance and operations after the initial PRA is completed.
- Procedure DPT-RS-00008 [53], Preparation and Maintenance of Unavailability Models, describes the process used to prepare, update and maintain unavailability models.
- Procedure DPT-RS-00003 [54], Evaluation of Risk Outside the Scope of the PRA, describes the process for assessing the nuclear safety risk of operating in a specific plant configuration, which is expected to be in place for a limited period of time, and which is not within the scope of the existing station PRA.
- Procedure DPT-RS-00004 [55], Risk Assessment of Proposed Changes to Engineering, Operations, Surveillance and Maintenance, describes the process used in performing risk assessments to evaluate proposed changes to engineering, operations, surveillance or maintenance programs, assess their acceptability with respect to Bruce Power safety goals and/or to the licensing requirements applicable to reliability, and identify vulnerabilities and means to lower risk as necessary.

From the PRA perspective, BP-PROC-00363 [44] is also implemented by the following procedures:

- DPT-RS-00002 [56], Risk Assessment of Operational Events, which prescribes how the risk of specific operational events should be evaluated.
- DPT-RS-00006 [57], Outage and Inage Risk Management, which describes the process to be used for Outage and Inage Risk Management with respect to Bruce Power safety goals and the licensing requirements applicable to reliability and risk.
- DPT-RS-00012 [58], Systems Important to Safety (SIS) Decision Methodology, which describes the logic and processes involved in evaluating the modelled systems in Bruce Power's PRAs, to determine which Systems Important to Safety are risk significant.
- DPT-NSAS-00011 [59], Configuration Management of Safety Analysis Software, which establishes the configuration management process for Safety Analysis Software including PRA analysis and applications, and scripts and utility codes. It is also designed for maintaining local software configuration of PRA software.
- DPT-NSAS-00013 [60], Guidelines for Managing Reference Data Sets, which describes the process of preparation, maintaining and usage of Reference Data Sets that are used in Safety Analysis Software including PRA.

Finally, three quality assurance related procedures also apply to PRA work and support the implementation of BP-PROC-00363 [44]:

- DPT-NSAS-00001 [61], Quality Assurance of Safety Analysis, which governs the quality assurance of safety analysis work in support of nuclear safety assessments;

- DPT-NSAS-00008 [62], Management of External Work for Nuclear Safety Analysis and Support, which prescribes how safety analysis work contracted to external parties should be managed; and
- DIV-ENG-00013 [63], Planning of Internal Work for Nuclear Safety Analysis, which prescribes how safety analysis work performed internally by Bruce Power should be planned.

The Bruce Power policies, programs and procedures that relate to PRA are identified in Table 4[2].

Table 4: Key Implementing Documents

Level 0	Level 1	Level 2	Level 3
BP-MSM-1: Management System Manual [41]	BP-PROG-10.01: Plant Design Basis Management [42]	BP-PROC-00363: Nuclear Safety Assessment [44]	DIV-ENG-00010: Probabilistic Risk Assessment Process [51]
			DPT-RS-00008: Preparation and Maintenance of Unavailability Models [53]
			DPT-RS-00004: Risk Assessment of Proposed Changes to Engineering, Operations, Surveillance and Maintenance [55]
			DPT-RS-00003: Evaluation of Risk Outside the Scope of the Probabilistic Risk Assessment [54]

[2] Table 4 lists the key governance documents used to support the assessments of the review tasks for this Safety Factor Report. A full set of current sub-tier documents is provided within each current PROG document. In the list of references, the revision number for the governance documents is the key, unambiguous identifier; the date shown is an indicator of when the document was last updated, and is taken either from PassPort, the header field, or the "Master Created" date in the footer.

Level 0	Level 1	Level 2	Level 3
			DPT-RS-00007: Preparation and Maintenance of Probabilistic Risk Assessments [52]
			DPT-RS-00002: Risk Assessment of Operational Events [56]
			DPT-RS-00006: Outage and Inage Risk Management [57]
			DPT-NSAS-00001: Quality Assurance of Safety Analysis [61]
			DPT-NSAS-00008: Management of External Work for Nuclear Safety Analysis and Support [62]
			DIV-ENG-00013: Planning of Internal Work for Nuclear Safety Assessment [63]
			DPT-NSAS-00011, Configuration Management of Safety Analysis Software [59]
			DPT-NSAS-00013, Guidelines for Managing Reference Data Sets [60]

Level 0	Level 1	Level 2	Level 3
	BP-PROG-11.01: Equipment Reliability [30]	BP-PROC-00778: Scoping and Identification of Critical SSCs [64]	DPT-RS-00012: Systems Important to Safety (SIS) Decision Methodology [58]

5. Results of the Review

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

5.1. Existing Probabilistic Safety Analysis

The existing PSA, including the assumptions used, the fault schedule, the representations of operator actions and common cause events, the modelled plant configuration and consistency with other aspects of the safety case were reviewed.

Within the Bruce B Risk Assessment (BBRA), the Bruce B PRA model is the result of a continuing process of updates and improvements that began in 1999 with the development of the original BBRA model by Ontario Hydro [65]. Since then, the Bruce B PRA and models have been updated when necessary to reflect the plant as built and operated. A full summary of the changes made to the BBRA model since its inception is provided in Appendix F of the Bruce B Risk Assessment Level 1 At-Power Model Integration Report including Emergency Mitigation Equipment (EME) [66].

The preparation of the Level 1 and Level 2 PRAs is based on the Bruce Power PRA guides for specific plant states and types of initiating events considered. These PRA guides are used to describe the technical details of the PRA methodology and serve as reference documents for PRA developers, practitioners and other knowledgeable stakeholders. A list of current Bruce B PRA analyses, and corresponding PRA guides, is given below.

1a) Level 1 At-Power Internal Events PRA [66] and [67];

1b) Level 1 At-Power Internal Events PRA Guide [68];

2a) Level 1 Outage Internal Events PRA [69];

2b) Level 1 Outage Internal Events PRA Guide [70];

3a) Level 2 At-Power Internal Events PRA [71];

3b) Level 2 At-Power Internal Events PRA Guide [72];

4a) Level 2 Outage Internal Events PRA [73];

5a) At-Power Internal Fire PRA [74],

5b) Internal Fire PRA Guide [75];

6a) At-Power Internal Flood PRA [76];

6b) Internal Flood PRA Guide [77];

7a) At-Power Seismic PRA [78];

7b) Seismic PRA Guide [79];

8a) At-Power High Wind PRA [80];

8b) High Wind PRA Guide [81].

In addition, Bruce Power has conducted and issued:

9a) External Hazards Assessments [82], [83], [84], [85], [86];

9b) External Hazards Screening and Disposition Guide [87].

The PRA guides used for the preparation of the Bruce B PRAs have been accepted for use by the CNSC, as documented in the letters [88], [89], [90], [91], [92] and [93]. The External Hazard guide and assessments were accepted by CNSC per the response letter [94].

The current Level 1 and Level 2 Bruce B PRAs take into consideration applicable multi-unit impacts. Throughout the update history of the BBRA model, as summarized in Appendix F of the Bruce B Risk Assessment Level 1 At-Power Model Integration Report including EME [66], continuing efforts have been made to improve its plant-specificity. Specific improvements include updating test interval frequencies, plant-specific changes of fault trees for selected systems, updated modelling of initiating events, inclusion of uncertainty data, implementation of Common Cause Failure (CCF) events, updating of the failure database and merging the Level 1 and Level 2 databases, revisions of Human Interaction (HI) failure probabilities, integration of the At-Power and Outage Level 1 system models, inclusion of EME into the Level 1 PRA, and merging the EME Fault Tree (FT) database into the BBRA database to form a single BBRA database for Level 1 and Level 2 At-Power models and Level 1 Outage model. Updates to the fault tree models are described in the latest PRA Level 1 At-Power reports [66] and [67].

The structure and analysis of event trees used in BBRA are based on the approach described in Chapter 4 of the Bruce B Probabilistic Risk Assessment Level 1 At-Power Event Tree Analysis for project B1130 [95]. Sections 4.2 to 4.26 of [95] describe the event tree (ET) analysis for each set of initiating events (IEs) considered, including a description of the IE, its ET diagram and functional fault trees, and associated assumptions. Section 4.27 of [95] provides a summary of the failure criteria for the various branch point mitigating systems.

All fault trees were developed in the Computer Aided Fault Tree Analysis (CAFTA) environment, using the standard fault tree development process and structure outlined in Section 2.3.5 of the Bruce Power Level 1 At-Power PRA Guide [68]. Updates to the fault tree models are described in the latest PRA Level 1 At-Power reports [66] and [67].

The treatment of CCFs in the current BBRA model is based on the methodology documented in [96]. The methodology has been accepted by the CNSC per [97].

The guidance on the methods used in the BBRA model for quantification of HI failure events is described in Section 2.4 of the Bruce Power Level 1 At-Power PRA Guide [68].

The CNSC conducted an inspection [98] of the Bruce Power Probabilistic Safety Assessment, and the specific focus was compliance of the Level 1 and Level 2 At-Power Internal Events PRAs with the requirements of CNSC S-294[3]. The inspection based its conclusions on examination of sample PRA scenarios. The results of the inspection were that Bruce Power has followed the CNSC accepted methodology for producing the PSA reports (as described in the Level 1 and Level 2 PRA Guides [68] and [72]), specifically in the areas of initiating events' quantification, uncertainty, sensitivity and importance analyses. It was also found that Bruce Power has a process for making changes to the PRA models as required by the S-294 standard, that the containment analysis in the Level 2 PRA and the interface between the Level 1 and Level 2 analyses are in agreement with the methodology, as are the definitions of the Plant Damage States (PDSs) and Release Categories (RCs).

However, the inspection also found that the updates of the fault tree analysis are not sufficiently traceable, and that some assumptions in FT models are not supported by the methodology, that quantification of HI events should be refined and should be applied consistently, and that the treatment of basic event reliability parameters does not fully reflect the plant as built and operated. As a result of these findings, eight (8) Action Notices and eleven (11) Recommendations were issued, as documented in the inspection report [98]. Bruce Power has submitted a response letter [99] describing specific corrective action plans that are being pursued to address the findings of the inspection. An Action Notice is defined in Appendix A of the CNSC Inspection Report [98] as "a written request that the licensee…take action to correct a non-compliance that is not a direct contravention of the NSCA, the applicable regulations, licence conditions, codes or standards, but that can compromise safety…and that may lead to a direct non-compliance if not corrected". Taking into account that the Action Notices do not reflect a direct non-compliance with codes and standards, it is therefore concluded that the findings of the CNSC inspection [98] do not result in additional gaps in this Safety Factor report. Further details on the inspection conclusions can be found in Section 7.3.

The requirements considered within the scope of this review element are assessed to be met.

5.2. Consistency of Accident Management Programs with PSA Models and Results

Accident management programs for accident conditions (design basis accident conditions and design extension conditions) and their consistency with PSA models and results were reviewed.

Bruce Power has issued a Severe Accident Management (SAM) program [100], which has been developed to deal with the possibility of a severe accident occurring on a single reactor unit operating initially at high power. Severe Accident Management Guidelines (SAMG) are currently being updated to implement improvements proposed in the CANDU Owners Group (COG) joint project JP4426 in response to the events at the Fukushima Daiichi plant [101]. The

[3] A detailed assessment of CNSC REGDOC-2.4.2, which superseded S-294, is included in Appendix B (B.1).

scope of the project includes responses for multi-unit and Irradiated Fuel Bay (IFB) events in severe accident conditions, and SAMG for shut down units or low-power operation.

As part of the current SAM program, Bruce Power has issued a number of SAMG documents, including a hierarchy of guides and procedures implementing the SAM procedure [100], under the Technical Support Group User's Guide [102]. The hierarchy defines conditions for entry into a SAM process, and it contains a structured set of SAM documentation, e.g., a Diagnostic Flow Chart [103], personnel instructions [104] [105] and a severe challenge status tree [106], including a set of specific procedures (i.e., severe accident guides (SAG), severe challenge guides (SCG), severe accident exit guides (SAEG), and computational aids (CA)). This documentation provides a pre-planned, systematic approach to guide the plant response in case of a severe accident.

The current SAMG has been developed by taking PRA results across the industry into account.

In this context PRA has and is being used to inform the SAMG program. The design basis accident management function is well represented in the current PRAs and is being extended to include credit for EME functions in the PRA. It is Bruce Power's position that SAMG credits beyond EME need not be incorporated into the PRA at this time. Bruce Power plans to retain the SAMG function as a residual risk management measure and not to credit it explicitly in PRA.

The requirements considered within the scope of this review task are assessed to be met.

5.3. Sufficiency of Scope and Applications of PSA

The scope and applications of the PSA were reviewed to determine if they are sufficient.

The sufficiency of the scope of a PRA can be judged on the basis of CNSC REGDOC-2.4.2 [23]. A clause-by-clause assessment of CNSC REGDOC-2.4.2 has been performed and is documented in Appendix B (B.1). In particular, Clause 4.1 of CNSC REGDOC-2.4.2 requires that a Level 1 and Level 2 PSA be performed for each NPP, with considerations including the reactor core and other radioactive sources such as the irradiated fuel bay, and taking into account multi-unit impacts.

The scope of the Bruce B PRA encompasses Level 1 and Level 2 analyses for the at-power and outage plant states, initiated by internal and external events. A full list of the current Bruce B PRAs is given in Section 5.1.

The main results of the Level 1 PRAs are frequencies of core damage states that can result from various accident sequences. The core damage states are defined based on their severity, time of progression and other features using insight from deterministic analyses. The frequencies of consequential core damage states are summed to obtain the Severe Core Damage Frequency (SCDF) for comparison against the corresponding safety goal.

The Level 2 PRAs further develop accident sequences from the Level 1 analyses, to obtain estimates of frequencies of radioactive releases outside of the reactor containment system. Release categories are defined based on their radioactive contents, duration and location of release, using deterministic analyses. The frequencies of specific release categories are summed to obtain estimates for comparison against the two safety goals associated with Level 2 PRA: Large Release Frequency (LRF) and Small Release Frequency (SRF).

As described in more detail in Section 5.5.1, the definition and quantitative values of the above safety goals, as used in the Bruce B PRA, are consistent with the requirements of Clause 4.2.2 of CNSC REGDOC-2.5.2 [39]. It is noteworthy that in the current regulatory framework the safety goals are defined on a per unit basis, whereas a definition of site-wide goals has in the past not been required in order to demonstrate adequate safety of the multi-unit station. Discussions on this topic between the industry and the CNSC are ongoing and Bruce Power will meet any new requirements resulting from these discussions. A proposed approach to site-wide characterization and assessment of Nuclear Power Plant risk can be found in the COG report [107]. Also, aspects of a whole-site PSA for CANDU reactors are the subject of COG JP 4499 [108].

Safety assessment of the irradiated fuel bay has been conducted outside the scope of PRA, as documented in [109]. This analysis was reviewed by CNSC and found acceptable [110].

In addition to the Level 1 and Level 2 internal events PRAs, several PRAs have been prepared for internal and external hazards, e.g., a fire PRA, a seismic PRA, an internal flooding PRA, and others. A full list of the current Bruce B PRAs is given in Section 5.1.

Bruce B PRAs are prepared and maintained under the general process described in the BP governing document [51], which establishes the requirements for the use of PRA at BP nuclear facilities. Within this framework, department procedure [52] provides instructions for the preparation and maintenance of plant-specific PRAs, defines the process for preparing a PRA as well as the systematic process of updating the PRA in order to maintain it as a "Living PRA".

In particular, regular updates of the BBRA model incorporate accumulated significant changes stemming from design, operational, maintenance, analysis and PRA applications experience, as required to keep the PRA consistent with the as built and as operated state of the plant. The process of periodic risk reassessment in PRA, as defined in the procedure [52], is based on changes from significant operational events, approved, committed, or implemented changes to engineering, operations, surveillance and maintenance (based on [55]), evaluations of risk outside the scope of the existing PRA (based on [54]), design changes and component reliability updates (based on the Annual Reliability Report [111]), issues from operating experience, etc. A process of continuous maintenance of the PRA model has been implemented by Bruce Power since the development of the original BBRA model in 1999. A full summary of updates of the BBRA model is given in Appendix F of the current Level 1 Internal Events document [66]. The development and implementation of the continuous and ongoing PRA maintenance process constitutes a strength as it exceeds the regulatory requirement of CNSC REGDOC-2.4.2 [23] that PRA model be updated every five years (requiring more frequent updates only if the facility undergoes major changes).

To support continued safe and reliable operation of the plant, Bruce Power intends to continue to maintain and update the BBRA for consistency with NPP testing and configuration management, taking into account the currently pursued asset management initiatives and associated with its risk-informed decision making practices.

The requirements considered within the scope of this review task are assessed to be met.

5.4. Status and Validation of Analytical Methods and Computer Codes Used in PSA

The status and validation of analytical methods and computer codes used in the PSA were reviewed.

Bruce B PRA models are built using analytical methods that are well-established in probabilistic risk analysis, as described in the Level 1 and Level 2 At-Power Internal Events PRAs [66], [67] and [71]. These methods include event trees to model accident progression sequences and fault trees to model failure probabilities of mitigating systems. In addition, Bayesian methodology is used in updating frequencies of initiating events. The methodology used to model CCFs was submitted to the CNSC [96], and was accepted by CNSC per Reference [97]. The basic event reliability models and the methods used for quantification of HI failure events used in the BBRA are described in Sections 2.5.2 and 2.4, respectively, of the Level 1 PRA Guide [68].

The current Level 1 and Level 2 BBRA models [66], [71] are implemented using a standard software package CAFTA, developed by Electric Power Research Institute (EPRI). The uncertainty and importance analyses have been performed using CAFTA-associated codes UNCERT and SYSIMP, and post-processing of fault trees has been done using a QRECOVER utility. In addition, the Modular Accident Analysis Program (MAAP) for CANDU, MAAP4-CANDU, is used in the Level 2 PRA [71] to assess the consequences of severe core damage progression challenging the containment system.

The computer codes for use in Level 1 and Level 2 PRAs have been submitted by Bruce Power for CNSC acceptance in letters [112], [113], [114], [115], and CNSC have accepted these codes in letters [116], [35], [117], and [118]. CNSC letter [35] also acknowledges that the standard CSA N286.7-99 [24] or equivalent QA computer code requirements are being followed by Bruce Power.

The requirements considered within the scope of this review task are assessed to be met.

5.5. Compliance with Relevant Probabilistic Safety Criteria

The results of the PSA were reviewed to determine if the PSA results show that risks are sufficiently low and well balanced for all postulated initiating events and operational states, and meet relevant probabilistic safety criteria.

5.5.1. Quantitative Safety Goals

The main results of the Bruce B PRA are reported as a comparison of the most important safety parameters with their respective Quantitative Safety Goals, which are numerical safety criteria to be used in association with PRA applications and against which the safety of nuclear reactors can be judged. The intent is to ensure the radiological risks arising from nuclear accidents associated with operation of nuclear reactors will be low in comparison to risks to which the public is normally exposed. Risk-based Safety Goals used in the Bruce B PRA to assess the acceptability of risk are defined in the Level 2 PRA Guide [72] for three safety parameters as follows:

- Quantitative Safety Goal for SCDF: Sum of frequencies of all event sequences that can lead to significant core degradation should not exceed 10^{-4} occurrences per reactor-year;
- Quantitative Safety Goal for SRF: Sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{15} Becquerels of Iodine-131 should not exceed 10^{-4} occurrences per reactor-year;
- Quantitative Safety Goal for LRF: Sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{14} Becquerels of Cesium-137 should not exceed 10^{-5} occurrences per reactor-year.

The results obtained in the Bruce PRA are summarized in Table 5, where the specific type of the PRA is identified in each row. These results individually meet all of Bruce Power's probabilistic safety goals. Note that, consistent with the requirements of Clause 4.2.2 of the CNSC REGDOC-2.5.2 [39], these safety goals are calculated on a per reactor (or per unit) basis (although multi-unit impacts on a single unit are considered). However, the results of the Bruce PRA cannot at present be compared with site-wide safety goals, as the latter have not been defined.

Clause 4.2.2 of CNSC REGDOC-2.5.2 [39] sets quantitative safety goals for aggregates of SCDF, SRF and LRF; namely, that the sum of SCDFs from all types of PRAs not exceed 10^{-5} occurrences per reactor-year, the sum of SRFs not exceed 10^{-6} occurrences per reactor-year, and the sum of LRFs not exceed 10^{-5} occurrences per reactor-year.[4] The guidance part of Clause 4.2.2 of CNSC REGDOC-2.5.2 [39] recommends that "calculations of the safety goals include all internal and external events as per CNSC REGDOC-2.4.2, PSA for Nuclear Power Plants", noting however that "aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated." As it can be seen from Table 5, each individual SCDF from the at-power internal events, outage internal events, internal flood, fire, seismic, and high-wind PRAs meets the Bruce Power safety goal of 10^{-4} occurrences per reactor-year defined in the PRA Guide [72]. Similarly, each individual LRF from the different PRA types meets the Bruce Power safety goal of 10^{-5} per reactor-year.

The sum of the individual SCDFs yields an aggregated SCDF of 2.49×10^{-5} occurrences per reactor-year, and the sum of the individual LRFs yields an aggregated LRF of 8.45×10^{-6} per reactor-year. Thus, the aggregated SCDF and LRF, obtained by summation across all Bruce B

[4] The requirements of CNSC REGDOC-2.5.2 [39] are intended for newly built NPPs. Consequently, the qualitative safety goals set in its Clause 4.2.2 are more stringent than those defined in the Bruce Power's PRA Guide [72]. Nevertheless, as explained in more detail in Section 3.3, CNSC REGDOC-2.5.2 is included in the scope of this PSR.

PRAs, do not meet the safety goals set in Clause 4.2.2 of CNSC REGDOC-2.5.2 [39]. This constitutes a gap against requirements for new NPPs (SF6-1).

5.5.2. Reliability of Systems Important to Safety

The guidance portion of Clause 7.6 of CNSC REGDOC-2.5.2 [39] states that "The design for reliability is based on meeting applicable regulatory requirements and industry standards. The design should provide assurance that the requirements of CNSC RD/GD-98, Reliability Programs for Nuclear Power Plants, will be met during operation. Not all Structures, Systems, and Components (SSCs) important to safety identified in the design phase will necessarily be included in the reliability program."

RD/GD-98 [25] provides requirements and guidance of the CNSC for the development and implementation of the reliability program of an NPP.

At a high level, the conditions for the availability of systems at Bruce B are set out in the Operating Policies and Principles document [43]. The Bruce Power Equipment Reliability program document BP-PROG-11.01 [30] identifies the high-level reliability procedures that map to each RD/GD-98 program requirement. The definition of SIS and the treatment of such systems in the context of PRA are described in the methodology document [52]. Risk significant systems are identified in the Level 1 and Level 2 At-Power Internal Events PRAs using the Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures. The ongoing record of reliability of SIS is documented in Annual Reliability Reports.

The 2015 Annual Reliability Report NK29-REP-09051.1-00016 [111] contains detailed results on the twelve Bruce B systems that comprise the SIS list. Quantitative unavailability models exist for nine of these systems; for others, CANDU Owners Group guidance COG-05-9011 [119] is followed, where the applicable initiating events frequencies are used as system monitoring parameters.

Clause 7.6 of CNSC REGDOC-2.5.2 [39] requires that, for newly built reactors, "the safety systems and their support systems shall be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 0.001". Per [111], all safety systems meet this requirement.

As per guidance provided by CNSC RD/GD-98, the resulting unavailabilities are assessed against their respective targets. The unavailability targets for the SIS were set out based on their design and operational requirements, per Section 2.3.2 of the COG guidance document COG-05-9011 [119].

In 2015, none of the twelve Systems Important to Safety exceeded their Bruce Power Predicted Future Unavailability (PFU) targets. Also per 2015 Annual Reliability Report [111], Actual Past Unavailability (APU) was observed for four out of twelve Systems Important to Safety. The four systems were Emergency Coolant Injection System, Emergency Water System, Shutdown System One and Shutdown System Two. The APU for the Emergency Water System was above its target. Events that caused the high APU have been addressed through Bruce Power's corrective action process. The report also mentioned that there were five missed and twenty-two deferred Safety System Tests, and zero missed and thirteen deferred Predefined

Maintenance items; the deferrals were evaluated using the BBRA and found to be acceptable based on system configuration and unavailability targets. Station Condition Records have been written to capture and trend these items.

The requirements considered within the scope of this review task are assessed to be met.

5.5.3. Reliability of the Shutdown Function

The guidance provided in clause 8.4.2 of REGDOC-2.5.2 [39] states that the reliability of the shutdown function should be such that the cumulative frequency of failure to shut down on demand is less than 10^{-5} failures per demand, and the contribution of all sequences involving failure to shut down to the large release frequency is less than 10^{-7}/yr.

The Level 1 PRA At-Power Model Integration Report including EME [66] incorporates all sequences including failure to shut down into the fuel damage category FDC1, whose value is estimated as 2.87E-8 occurrences per reactor per year. Thus the guidance target of cumulative frequency of failure to shut down on demand being less than 10^{-5}/yr is demonstrated by the fuel damage category FDC1 in the Level 1 PSA.

Per Level 2 At-Power Summary Report [71], from Level 1 PRA the Fuel Damage Category 1 (FDC1) represents all sequences involving rapid accident progression resulting from failures to shut down the reactor when required. FDC1 is conservatively assumed to cause early consequential containment failure with a 0.5 probability and the failure sequence is assigned to a unique PDS, i.e. PDS1. Release Categories (RCs) are defined to bin the consequences associated with containment event tree end-states to facilitate comparison with safety goals. RC0 consists of single unit events (PDS1), two-unit events (PDS3A) and three- or four-unit events (PDS3). The contributions to RC0 of PDS3 and PDS3A are 94% and 4%, respectively. That means the contribution of PDS1 to RC0 is approximately 2%. The frequency of RC0 is included in the LRF calculation. Per Table 4 of [71], RC0 frequency is 4.71E-6, which means that the contribution to it from PDS1 is 9.42E-8. This is below the target for the contribution of all sequences involving failure to shut down to the large release frequency of the safety goals of 10^{-7}/yr.

The requirements considered within the scope of this review task are assessed to be met.

Table 5: Summary of Safety Parameters Obtained in Bruce B PRA

Type of PRA	Document References	Values of Safety Parameter(s) [per reactor year]	Notes
Level 1 At-Power Internal Events	[66]	SCDF=5.18E-06	PRA including EME

Type of PRA	Document References	Values of Safety Parameter(s) [per reactor year]	Notes
Level 1 Outage Internal Events	[69]	SCDF=8.30E-06	
Level 2 At-Power Internal Events	[120]	LRF=6.93E-07 SRF=7.14E-07	PRA including EME (Ref. [120])
Level 2 Outage Internal Events	[73]	Not Applicable	No results for LRF or SRF are presented. The March 2014 CNSC submission [73] states that there is no need to complete a detailed Level 2 Outage Internal Events PRA, justifying this by the acceptably low SCDF from the Level 1 Outage Internal Events PRA and by a limited Level 2 Outage study.
Levels 1&2 At-Power Internal Fire	[121]	SCDF=4.06E-06 LRF=8.74E-07	PRA including EME; see Attachment A, item (e) of [121].
Level 1 At-Power Internal Flood	[76] [121]	SCDF=4.60E-07 LRF<1E-06	For SCDF see [76], and Attachment B, Table 2 of [121]. For LRF, it is stated in [76] (Enclosure 13) that since the SCDF is <1E-06, the LRF must also be <1E-06, hence no Level 2 PRA for internal flood was performed.

Type of PRA	Document References	Values of Safety Parameter(s) [per reactor year]	Notes
Levels 1&2 At-Power Seismic	[78]	SCDF=7.20E-07, for events with return frequency up to 10000 years – equivalent to the Review Level Earthquake LRF=7.20E-07	PRA including EME [78] See Attachment B, Table 4 of [121]. The Containment Failure Frequency (CFF) = 2.80E-06 is provided in [78].
Levels 1&2 Outage Internal Fire	Not Applicable	Not Applicable	The January 2014 CNSC submission [122] provides justification why outage PRA for internal fires, seismic events and internal floods does not need to be performed for S-294 compliance.[5]
Levels 1&2 Outage Internal Flood	Not Applicable	Not Applicable	The January 2014 CNSC submission [122] provides justification why outage PRA for internal fires, seismic events and internal floods does not need to be performed for S-294 compliance. [5]
Levels 1&2 Outage Seismic	Not Applicable	Not Applicable	The January 2014 CNSC submission [122] provides justification why outage PRA for internal fires, seismic events and internal floods does not need to be performed for S-294 compliance.[5]

[5] In Reference [122], Bruce Power requests CNSC's agreement for this exclusion on the basis that "the risk from internal fires, internal floods and seismic events for a single unit on an outage is both low and well-managed in accordance with the principle that the level of detail in the PRA should be consistent with the level of risk". Technical details supporting this reasoning are provided in the appendices to [122]. The CNSC has accepted the arguments in [122] to exclude internal fires, internal floods and seismic events from the scope of Bruce Power's outage PRA [123].

Type of PRA	Document References	Values of Safety Parameter(s) [per reactor year]	Notes
Levels 1&2 At-Power High Wind	[80]	SCDF=6.16E-06 LRF<6.16E-06 (bounded by SCDF)	PRA including EME High wind PRA is done for site rather than for individual units.
Levels 1&2 External Flooding (and other External Hazards)	[82] [83] [84] [85] [86]	Not Applicable	PRAs have not been done for these external hazards. But hazards assessments have been performed in accordance with the External Hazards Screening and Disposition Guide [87] and documented in references [82], [83], [84], [85], [86].

5.6. Sufficiency of Scope and Application of PSA in Assessing Proposed Improvement Options

The existing scope and application of the PSA were reviewed to determine whether they are sufficient for its use to assist the PSR global assessment, for example, to compare proposed improvement options.

In laying out general recommendations for a Periodic Safety Review, IAEA SSG-25 [40] states in Clause 2.17 that "in order to integrate the results of the reviews of individual Safety Factors, the operating organization should perform a global assessment of safety at the plant. The global assessment should consider all findings and proposed improvements from the Safety Factor reviews and interfaces between different Safety Factors". It is further stated in Clause 4.22 that "the level of plant safety should be determined by a global assessment reflecting, among other things, the combined effects of all Safety Factors. It is possible that a negative finding (deviation) in one Safety Factor can be compensated for by a positive finding (strength) in another Safety Factor".

In this context, one of the important features of a PRA indicating its sufficiency to assist a Global Assessment Review (GAR) is the degree to which the PRA facilitates clear interfaces with safety aspects assessed in Safety Factors other than the current one. The interfaces discussed in Section 6 indicate that the contents of the Bruce B PRA should incorporate information on the actual state of the plant (which may include design and the reliability program), should be consistent with and supported by results of deterministic safety analyses and hazard assessments. The extent to which these aspects are reflected in the Bruce B PRA is discussed in more detail in Sections 5.3 and 5.5.2.

According to the Bruce B PSR Basis Document [5], the findings of the Safety Factor reviews will be consolidated and integrated in a GAR to arrive at overall conclusions regarding the continued safe operation and major component replacement in Bruce B. The GAR will also identify potential improvement opportunities that would address gaps between the current plant design and operation and modern codes, standards and practices, and describes how these opportunities are consolidated, ranked, and prioritized.

PRA could be used to assess the risk of potential improvement options by revising the PRA to take into account design changes and determining resulting risk increase or risk decrease. For example, existing Fussell-Vesely importance measures could be considered for evaluation of potential physical improvements.

The requirements considered within the scope of this review task are assessed to be met.

6. Interfaces with Other Safety Factors

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

- "Safety Factor 1: Plant Design" performs a review against REGDOC-2.5.2, which includes requirements for PSA and relevant safety goals (e.g., severe core damage frequency).
- "Safety Factor 3: Equipment Qualification" in Sections 4 and 5.2, verifies that programs exist which identify and categorize equipment to monitor and maintain it appropriately for the life of the plant.
- "Safety Factor 5: Deterministic Safety Analysis" in Section 5.1, reviews analysis methods and computer codes used for deterministic safety analysis in comparison with current standards and requirements including CSA N286.7-99 [24].
- "Safety Factor 7: Hazards Analysis" in Sections 5.1 and 5.2, respectively, assesses the systematic identification of external and internal hazards, some of which are PRA's Postulated Initiating Events (PIEs).

7. Program Assessment and Adequacy of Implementation

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

- Self-Assessments;
- Internal and External Audits and Reviews;
- Regulatory Evaluations; and

- Performance Indicators.

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

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

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

7.1. Self-Assessments

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

The self-assessments:

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

A review of audits and inspections that could potentially be relevant to this Safety Factor revealed that there have not been any PRA-related FASAs in the last five years.

7.2. Internal and External Audits and Reviews

The objective of the audit process as stated in Section 4.2 of BP-PROG-15.01 [124] 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 [26]).

There have not been any audits or reviews in the last five years relevant to this Safety Factor.

7.3. Regulatory Evaluations and Reviews

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

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

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

The CNSC conducted a Type II compliance inspection of the Bruce Power Probabilistic Safety Assessment in September 2014. The objective of the inspection was to verify that the submitted PSA followed the accepted methodologies in accordance with S-294 [97].

Based on the inspection sample reviewed, Bruce Power has followed the methodology for producing the PSA reports, specifically in the areas of Initiating Events, Event Trees, Uncertainty, Sensitivity and Importance Analysis in Level 1 PSA as well as for Level 2 PSA. Gaps have been identified for Fault Trees and data analysis which did not follow the methodology. For human reliability analysis, deficiencies were found in the consistent application of the methodology.

As a result of this inspection, 8 Action Notices and 11 Recommendations were raised which are documented in BRPD-AB-2014-012 - Probabilistic Safety Assessment Inspection [97]. As documented in the response letter [99], Bruce Power has developed and is currently pursuing specific corrective action plans that address all of the findings of the inspection.

The action notices and recommendations made in the CNSC inspection [97] have been reviewed in light of the assessments performed as part of this Safety Factor Review to ensure that these findings do not result in gaps in addition to those listed in Table 6. Namely, according to the definitions given in the CNSC Inspection Report [97], a Recommendation is less consequential than an Action Notice, with the latter being defined as “a written request that the licensee…take action to correct a non-compliance that is not a direct contravention of the NSCA, the applicable regulations, licence conditions, codes or standards, but that can compromise safety…and that may lead to a direct non-compliance if not corrected”. Since the definition states that an Action Notice does not reflect a direct non-compliance with codes and standards, it is therefore concluded that, for the purposes of the present report, the findings of the CNSC Inspection [97] do not result in additional gaps. This conclusion is consistent with the general finding of the Inspection that Bruce Power “meets the regulatory requirements, with the exception of the above-noted non-compliances with Bruce PRA procedures. CNSC staff did not find evidence of unsafe operation that would result in undue risk to the health and safety of persons, the environment, or that would compromise respect for Canada’s international obligations”.

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.

There are no specific performance indicators associated with probabilistic safety assessment that are currently used.

The CNSC Staff Integrated Safety Assessment of Canadian Nuclear Power Plants for 2014, issued in September 2015 [125], summarizes the 2014 ratings for Canada’s NPPs in each of the 14 CNSC Safety and Control Areas (SCAs), including safety analysis (which itself includes PSA). For 2014, the Bruce B rating for the safety analysis SCA was “satisfactory”.

8. Summary and Conclusions

The overall objectives of the Bruce B PSR are to conduct a review of Bruce B against modern codes and standards and international safety expectations, and to provide input to a practicable set of improvements to be conducted during the MCR in Units 5 to 8, as well as U0B, and during asset management activities to support ongoing operation of all four units, that will enhance safety to support long term operation. The specific objectives of the review of this Safety Factor are to determine:

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

These specific objectives have been met by the completion of the review tasks specific to probabilistic safety analysis.

One strength was identified in the Periodic Safety Review of Safety Factor 6, as follows:

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

Table 6 summarizes the key issue arising from the Periodic Safety Review of Safety Factor 6.

Table 6: Key Issues

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

The overall conclusion is that, with the exception noted in Table 6, Bruce Power meets the requirements of the Safety Factor related to the Probabilistic Safety Analysis.

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[108] COG-JP-4499-001-R000, CANDU Whole-Site Probabilistic Safety Assessment Project Execution Plan, CANDU Owners Group Report, December 2014.

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

No codes or standards relevant to Safety Factor 6 were subjected to high-level assessment. This Appendix is retained only for consistency with the Appendix numbering scheme in all other Safety Factor Reports.

Appendix B – Clause-By-Clause Assessments Against Relevant Codes and Standards

This appendix presents the clause-by-clause assessments that are performed for this Safety Factor. The PSR Basis Document provides the following compliance categories and definitions for clause-by-clause assessments:

- Compliant (C) – compliance has been demonstrated with the applicable clause;
- Indirect Compliance (IC) – Compliance has been demonstrated with the intent of the applicable clause;
- Acceptable Deviation (AD) – Compliance with the applicable clause cannot be demonstrated; however, a technical assessment has determined that the deviation is acceptable. For this case a detailed discussion and explanation shall be included in the PSR documentation;
- Gap – system design and/or operational improvements may be necessary;
- Guidance: A potential programmatic, engineering, analytical or effectiveness gap found against non-mandatory guidance;
- Relevant but not Assessed (RNA) – The PSR 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 also includes the guidance portion of clauses in which a gap has already been identified against the requirement;
- Not Relevant (NR) – The topic addressed in the specific clause is not relevant to the safety factor under consideration but may well be assessed under a different Safety Factor; and
- Not Applicable (NA) – The text is not a clause that provides requirements or guidance. Also used if the clause does not apply to the specific facility.

B.1. CNSC REGDOC-2.4.2, Probabilistic Safety Analysis

In support of the review tasks listed in Section 5, a detailed assessment of the CNSC REGDOC-2.4.2 requirements has been performed in Table B1.

Table B1: CNSC REGDOC-2.4.2, Probabilistic Safety Analysis

Article No.	Clause Requirement	Assessment	Compliance Category
4.1	Perform a level 1 and level 2 PSA for each NPP. Considerations shall include the reactor core and other radioactive sources such as the spent fuel pool (also called irradiated fuel bay). Multi-unit impacts, if applicable, shall be included. For radioactive sources outside the reactor core, the licensee may, with the agreement of persons authorized by the Commission, choose an alternate analysis method to conduct the assessment.	The Bruce B Probabilistic Risk Assessment (PRA) (synonymous with Probabilistic Safety Assessment) includes Level 1 and Level 2 analyses. The Bruce B PRA model, abbreviated as BBRA, is the result of a continuing process of updates and improvements that began in 1999 with the development of the original BBRA model by Ontario Hydro. Since then, the Bruce B PRA and models have been updated to reflect the plant as built and operated and as required. A full summary of the changes made to the BBRA model since its inception is provided in Appendix F of the Bruce B Risk Assessment (BBRA) Level 1 At Power Model Integration Report including Emergency Mitigation Equipment (EME), B1401/RP/004 R01, July 18, 2014 (see also Enclosure 11 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014). The main results of the Level 1 PRAs are frequencies of core damage states that can result from various accident sequences. The core damage states are defined based on their severity, time of progression and other features using insight from deterministic analyses. The frequencies of most consequential	C

Article No.	Clause Requirement	Assessment	Compliance Category
		core damage states are summed up to obtain the safety goal of severe core damage frequency (SCDF). Bruce Power has developed PRAs for at-power and outage plant states, with contributions from internal initiating events (IEs), from internal hazards and from external hazards. The Level 2 PRAs further develop accident sequences from the Level 1 analyses, to obtain estimates of frequencies of radioactive releases outside of the reactor containment system. Release categories are defined based on their radioactive contents, duration and location of release, using deterministic analyses. The frequencies of specific release categories are summed up to obtain estimates of the two safety goals associated with Level 2 PRA: Large Release Frequency (LRF) and Small Release Frequency (SRF). The current Level 1 and Level 2 Bruce B PRAs are plant specific. They also take into consideration applicable multi-unit impacts (see assessment of clause 4.3 for more details). The latest Bruce B Level 1 and Level 2 PRA updates have been submitted to the Canadian Nuclear Safety Commission (CNSC) and are documented as follows: 1. 2013 Bruce B PRA Level 1 At-Power Summary Report, B1294/RP/002 R01, August 23, 2013 [NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk,	

Article No.	Clause Requirement	Assessment	Compliance Category
		December 24, 2013] 2. 2013 Bruce B Level 2 At-Power Internal Events Risk Assessment Summary Report, B0900/RP/055 R01, December 2013 [NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] 3. 2014 Bruce B Risk Assessment (BBRA) Level 1 At-Power PRA Model Integration Report (including Emergency Mitigation Equipment (EME)), B1401/RP/004 R01, July 18, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] 4. 2014 Bruce B Risk Assessment (BBRA) Level 1 Outage PRA Model Summary Report, B1401/RP/002 R01, July 17, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] The latest Bruce B external events PRA/assessment updates have been submitted to the Canadian Nuclear Safety Commission (CNSC) and are	

Article No.	Clause Requirement	Assessment	Compliance Category
		documented as follows: 1. K-410003-REPT-0074-R001, Seismic Probabilistic Risk Assessment (PRA) Summary Report, July 24, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014] 2. B1401/RP/006 R001, High Wind PRA Model Report, July 18, 2014 [see Enclosure 9 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] 3. K-449958-REPT-0007-R003, External Hazards Assessment - Final Report, July 24, 2014 [see Enclosure 7 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] In NK21-CORR-00531-11091/NK29-CORR-00531-11491, Bruce Power concludes that there is no need to complete a Level 2 outage PRA on the basis of the following: • Level 1 Outage PRA results characterize the risks encountered during outages. The results satisfy the Core Damage Frequency (CDF) safety goal with good margin; • Level 2 At-Power PRA results and insights	

Article No.	Clause Requirement	Assessment	Compliance Category
		provide significant information on the behaviour of containment during a core damage event and effectively bound the contribution of outage unit accidents to release scenarios; • Modular Accident Analysis Program (MAAP) analysis undertaken specifically for the outage state show that only Drained Guaranteed Shutdown State sequences have the potential for large release. If the CDF of this state is conservatively assumed to lead directly to LRF, the safety goal is still met. The result of this bounding assessment obviates the need for further outage Level 2 considerations. A limited PRA Level 2 outage study has also been performed in support of the above: Bruce Nuclear Generating Station (NGS) A and B Level 2 Outage Probabilistic Risk Assessment, B1380/RP/001 R01, September 13, 2013 (see also Enclosure 1 of NK21-CORR-00531-11017/NK29-CORR-00531-11413, dated January 22, 2014). Regarding the consideration of radioactive sources outside the reactor core, a safety assessment of the irradiated fuel bay has been conducted outside the scope of PRA, as documented in [NK21-CORR-00531-10341/NK29-CORR-00531-10750, Bruce Power Irradiated Fuel Bay Structural Integrity Analysis, Bruce Power Letter, F. Saunders to R. Lojk,	

Article No.	Clause Requirement	Assessment	Compliance Category
		March 26, 2013]. This analysis was reviewed by the CNSC and found acceptable as per their response letter [NK21-CORR-00531-10565/NK29-CORR-00531-10965, CNSC Review of Bruce Power's Irradiated Fuel Bay Structural Integrity Analysis (Fukushima Action Items 1.5.1, 1.6.1 and 1.6.2), CNSC Letter, R. Lojk to F. Saunders, June 3, 2013]. Multi-unit impacts are included in the Bruce Power PRA. As per 2013 Bruce B Level 2 At-Power Internal Events Summary Report, B0900/RP/055 R01, December 2013, initiating events are postulated to lead to severe accident progression at multiple units either because all units are challenged simultaneously by the event (loss of offsite power) or because the environmental impacts of high temperature steam from an event at one unit and subsequent condensation in the powerhouse impact the electrical systems of other units. These outcomes and related probabilities are determined by the Level 1 PRA but their impacts only become apparent at Level 2. Examples of initiating events with contribution to the severe core damage frequency are the large steam line break in an adjacent unit and the loss of forebay initiating events; see 2013 Bruce B PRA Level 1 At-Power Summary Report, B1294/RP/002 R01, August 2013. Modelling of multiple unit sequences in the Level 2 Internal Events PRA (see B0900/RP/055 R01, December 2013) is approximated by scaling the common containment volumes by a factor of two or four, such that the containment pressure response reflects the relative rate of energy generation and	

Article No.	Clause Requirement	Assessment	Compliance Category
		absorption from failure of two or four units. It is noteworthy that in the current regulatory framework the safety goals are defined on a per unit basis, whereas a definition of site-wide goals has in the past not been required in order to demonstrate adequate safety of the multi-unit station. Discussions on this topic between the industry and the CNSC are ongoing and Bruce Power will meet any new requirements resulting from these discussions. A proposed approach to site-wide characterization and assessment of Nuclear Power Plant risk can be found in the COG report COG 13-9034-R000, February 2014. Also, aspects of a whole-site PSA for CANDU reactors are the subject of COG JP 4499.	
4.2	Conduct the PSA under the management system or quality assurance program established in the licensing basis. Guidance: The CSA N286 management system requirements standard and CSA N286.7, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants, are referenced in the licensing basis of operating nuclear power plants. The PSA should be developed in a manner that is consistent with the management system.	Within the organization of Bruce Power's programs and processes, probabilistic safety analysis falls under the broader function of Nuclear Safety Assessment, which also covers activities such as deterministic safety analysis and criticality safety assessment. The Nuclear Safety Assessment function, together with the Design Management Function, falls under Bruce Power's Plant Design Basis Management Program. Nuclear safety is addressed at the highest level of the hierarchy in the Management System Manual BP-MSM-1-R012, Bruce Power, June 23, 2014. Bruce Power's Management System addresses and incorporates the following principle, consistent with Canadian Standards Association industry developed standard CSA N286-05, Management System Requirements	C

Article No.	Clause Requirement	Assessment	Compliance Category
		for Nuclear Facilities: • Safety is the paramount consideration guiding decisions and actions. The Management System also addresses and incorporates the following principles, consistent with CSA N286-05, Management System Requirements for Nuclear Power Plants: • The business is defined, planned and controlled • The organization is defined and understood • Personnel are competent at the work they do • Personnel know what is expected of them • Work is planned • Experience is sought, shared and used • Information is provided in time to the people who need it • The performance of work is controlled • The preparation and distribution of documents are controlled • Work is verified to confirm that it is correct • Problems are identified and resolved • Changes are controlled • Records are maintained • Assessments are performed An implementation strategy for the CSA N286-12 is in progress to be submitted to the CNSC by the end of 2016. CNSC staff have stated that in their view the CSA N286-12 version of CSA N286 “does not	

Article No.	Clause Requirement	Assessment	Compliance Category
		represent a fundamental change to the current Bruce Power Management System" and have acknowledged that "the new requirements in CSA N286-12 are already addressed in Bruce Power's program and procedure documentation". Furthermore, 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. The MSM governs the Plant Design Basis Management program (BP-PROG-10.01-R009). This program authorizes the use of the BP-PROC-00363-R003 Nuclear Safety Assessment procedure, which defines the elements, functional requirements, implementing procedures and key responsibilities associated with the Nuclear Safety Assessment (NSA) process. The BP-PROC-00363-R003 procedure satisfies relevant statutory, regulatory and licensing requirements from the CSA N286-05, Management System Requirements, and CSA N286.7, Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants. A component of the nuclear safety assessment, the implementation of procedure DIV-ENG-00010-R000, Probabilistic Risk Assessment Process, provides general governance of all PRA related procedures, establishes the process by which PRA is carried out, and provides Bruce Power risk based nuclear safety	

Article No.	Clause Requirement	Assessment	Compliance Category
		goals. The lower level procedures most relevant to PRA are listed below: • DPT-RS-00008-R000, Preparation and Maintenance of Unavailability Models • DPT-RS-00004-R001, Risk Assessment of Proposed Changes to Engineering, Operations, Surveillance and Maintenance • DPT-RS-00003-R001, Evaluation of Risk Outside the Scope of the PRA • DPT-RS-00007-R001, Preparation and Maintenance of Probabilistic Risk Assessments • DPT-RS-00002-R000, Risk Assessment of Operational Events • DPT-RS-00006-R001, Outage and Inage Risk Management • DPT-RS-00012-R001, Systems Important to Safety (SIS) Decision Methodology • DPT-NSAS-00011-R004, Configuration Management of Safety Analysis Software • DPT-NSAS-00013-R003, Guidelines for Managing Reference Data Sets • DPT-NSAS-00001-R006, Quality Assurance of Safety Analysis • DPT-NSAS-00008-R004, Management of External Work for Nuclear Safety Analysis and Support	

Article No.	Clause Requirement	Assessment	Compliance Category
		• DIV-ENG-00013-R001, Planning of Internal Work for Nuclear Safety Analysis Furthermore, specific PRA guides have been developed and are used to describe the technical details of the PRA methodology and serve as reference documents for PRA developers, practitioners and other knowledgeable stakeholders. These detailed PRA guides are: • Level 1 At-Power Internal Events PRA Guide • Level 1 Outage Internal Events PRA Guide • Level 2 At-Power Internal Events PRA Guide • Internal Flood PRA Guide • Internal Fire PRA Guide • Seismic PRA Guide • High Wind PRA Guide • External Hazards Screening and Disposition Guide As described in the above management system structure and hierarchy, the PRA is conducted under the management system and quality assurance program established in the licensing basis.	
4.3	The PSA models shall reflect the plant as built and operated (including multi-unit impacts), as closely as reasonably achievable within the limitations of PSA technology, and consistent with the risk impact.	The Bruce B PRA model described in the Bruce B Risk Assessment (BBRA) Level 1 At-Power Model Integration Report including Emergency Mitigation Equipment (EME), B1401/RP/004 R01, July 18, 2014, [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31,	AD

Article No.	Clause Requirement	Assessment	Compliance Category
		2014] adequately reflects specifics of the plant configuration and operation. The Bruce B systems' design, operation and testing is modelled in BBRA using a set of system fault trees specific to Bruce B. The plant-specificity of the BBRA model has been improving in the course of its multiple updates, carried out since its inception under the governance of DPT-RS-00007-R001, as summarized in Appendix F of the Bruce B Risk Assessment (BBRA) Level 1 At Power Model Integration Report including Emergency Mitigation Equipment (EME), B1401/RP/004 R01, July 18, 2014. In particular, one integrated database was created by combining databases initially developed for the Level 1 At-Power, Level 2 At-Power and Level 1 Outage PRAs. Other updates included revisions of the component failure rates, addition of probability parameters for maintenance and testing outage events, conditioning events and developed events, and updates of the frequencies of initiating events (IEs). The updates of frequencies of IEs are done using Bayesian techniques whereby distributions of frequencies (extracted from generic industry-wide data) are updated by taking into account CANDU-specific and Bruce B-specific operating experience. This methodology is described in the Level 1 At-Power Internal Events PRA Guide B-REP-03611-00005-R001.	

Article No.	Clause Requirement	Assessment	Compliance Category
		Application of Bayesian techniques to updating component failure rates is described in the Level 1 At-Power PRA Guide B-REP-03611-00005-R001. The Bayesian technique is based on the use of both the “prior knowledge” and the plant-specific data in deriving the failure rates. In order to apply the Bayesian approach to failure rates, prior distributions were obtained from the latest industry generic data sources and the plant-specific data were obtained from the latest Bruce NGS operating experience. Using these two sources, the BBRA failure rates underwent a major update in 2011. In addition, Bruce Power has been improving quantification of key screening human interaction (HI) error values based on importance, and completed the qualification of the Bruce B MAAP4-CANDU parameter file for severe accident analysis. Multi-unit impacts are included in the Bruce Power PRA. As per 2013 Bruce B Level 2 At-Power Internal Events Risk Assessment Summary Report, B0900/RP/055 R01 (NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013), December 2013, initiating events are postulated to lead to severe accident progression at multiple units either because all units are challenged simultaneously by the event (loss of offsite power) or because the environmental impacts of high temperature steam from an event at one unit and subsequent condensation in the powerhouse	

Article No.	Clause Requirement	Assessment	Compliance Category
		impact the electrical systems of other units. These outcomes and related probabilities are determined by the Level 1 PRA but their impacts only become apparent in the Level 2. Examples of initiating events with contribution to the SCDF are the large steam line break in an adjacent unit and the loss of forebay initiating events; see 2013 Bruce B PRA Level 1 At-Power Summary Report, B1294/RP/002 R01, August 2013 (NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power letter, F. Saunders to R. Lojk, December 24, 2013). Modelling of multiple unit sequences in the Level 2 Internal Event PRA (see B0900/RP/055 R01, December 2013) is approximated by scaling the common containment volumes by a factor of two or four, such that the containment pressure response reflects the relative rate of energy generation and absorption from failure of two or four units. Emergency Mitigating Equipment (EME) to be installed under the Fukushima Action Plan has been modelled, which includes modifications of event trees, incorporation of EME-related fault trees, databases and HI events into the following Bruce B PRAs: • Level 1 PRAs for At-Power and Outage Internal Events: B1294/RP/002 R01, NK29-03611.1 P NSAS, July 18, 2014 and B1401/RP/002 Rev 1, July 17, 2014 (NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Final	

Article No.	Clause Requirement	Assessment	Compliance Category
		Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014), respectively, and • PRAs for Seismic (K-410003-REPT-0074, R001, July 24, 2014), Fire (K-410003-REPT-0037, Rev 01, July 24, 2014), and High Wind Hazards (B1401/RP/006 R01, July 18, 2014).(see Enclosures 4, 6, and 9 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014) The CNSC conducted an inspection (see NK21-CORR-00531-11710/NK29-CORR-00531-12099, Action Item 2014-07-5551: CNSC Type II Compliance Inspection Report: BRPD-AB-2014-012 - Probabilistic Safety Assessment Inspection, CNSC Letter, K. Lafrenière to F. Saunders, November 6, 2014) of the Bruce Power Probabilistic Safety Assessment, whose specific focus was compliance of the Level 1 and Level 2 At-Power Internal Events PRAs with the requirements of CNSC S-294. The inspection found that Bruce Power followed CNSC accepted methodology on quantification of initiating events and in the event tree analysis. It was further found that Bruce Power has a process for making changes to the PRA models as required by the S-294 standard, that the uncertainty, sensitivity and importance analyses follow the accepted methodology, that the containment analysis in the Level 2 PRA and the interface between the Level 1 and Level 2 analyses are in agreement with the methodology, as are the	

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		definitions of the Plant Damage States (PDSs) and Release Categories (RCs). However, the inspection also found several shortcomings in the implementation of the PRA governance. Namely, it was pointed out that the updates of the fault tree analysis are not sufficiently traceable, and that some assumptions in FT models are not supported by the methodology, that quantification of HI events should be refined and should be applied consistently, and that the treatment of basic event reliability parameters do not fully reflect the plant as built and operated. As a result of these findings, 8 Action Notices and 11 Recommendations from the CNSC inspection were issued, as documented in the inspection report (NK21-CORR-00531-11710/NK29-CORR-00531-12099). Bruce Power has responded to the CNSC inspection and made a number of commitments to address the Action Notices and Recommendations - see NK21-CORR-00531-11721/NK29-CORR-00531-12110, CNSC Type II Compliance Inspection Report: BRPD-AB-2014-012 - Probabilistic Safety Assessment, Bruce Power Letter, F. Saunders to K. Lafrenière, January 16, 2015. These findings do not constitute a gap for the purposes of this review, because according to the definitions given in the CNSC Inspection Report, an Action Notice is "a written request that the licensee…take action to correct a non-compliance that is <u>not a direct contravention of the NSCA, the applicable regulations, licence conditions, codes or</u>	

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		standards, but that can compromise safety…and that may lead to a direct non-compliance if not corrected". Therefore, an acceptable deviation is assessed against this REGDOC 2.4.2 clause.	
4.4	Update the PSA models every five years. The models shall be updated sooner if the facility undergoes major changes. Guidance: Update the PSA models so that they adequately represent the as-operated plant conditions.	Current practice at Bruce Power is to continuously maintain the at-power BBRA model throughout the year and issue a reference model and summary update document for PSA applications approximately once a year. This follows the procedure described in Section 4.4.2 of DPT-RS-00007-R001, Preparation and Maintenance of Probabilistic Risk Assessments and ensures that the BBRA model is representative of the actual plant configuration and operation and testing at the station. The development and implementation of the continuous PRA maintenance process constitutes a strength as it exceeds the regulatory requirement. A full summary of the changes made to the BBRA model since its inception is provided in Appendix F of the Level 1 At-Power Model Integration Report, B1401/RP/004 R01, July 18, 2014 (NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014). The continuous model development is now viewed as implementation of the concept of "Living PRA", as defined in DIV-ENG-00010-R000 and DPT-RS-00007-R001: "Living PRA is a PRA/unavailability model that is re-	C

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		evaluated and updated periodically to reflect plant-specific design, operational and component reliability data changes. Design and operational changes to modeled systems requires revision of the specific PRA/unavailability models. Plant-specific nuclear power plant component reliability data is collected, evaluated and input back into the PRA based unavailability models on a frequent basis (i.e., typically yearly). The routine collection, evaluation and inputting of component reliability data ensures that the PRA/unavailability models are calculating risks/unavailabilities which are representative of changes in component reliability data over the life of the nuclear power plant." Bruce Power intends to continue to maintain and update BBRA for consistency with NPP testing and configuration management, taking into account the currently pursued asset management initiative and associated with its risk-informed decision making practices.	
4.5	Ensure the PSA models are developed using assumptions and data that are realistic and practical and, where required, supported by deterministic safety analysis or engineering assessments.	The original BBRA model assumptions were made based on the best available plant information and the best judgment of plant engineers at the time of its issue in 1999. Where there existed a lack of information, high degree of uncertainty or where inordinate resource effort may have been required, some conservatism in methodology and assumptions was used. Since then, as part of the existing BBRA maintenance process and updates, as summarized in Appendix F of the Bruce B Level 1 At Power PRA Report including Emergency Mitigation Equipment (EME), B1401/RP/004 R01, July 18, 2014 (NK21-	C

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		CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014) the models are being refined and updated using more realistic assumptions and data, as the assumptions are challenged through various plant risk applications. To achieve a realistic and up-to-date plant representation, the component failure database has been regularly revised in the course of BBRA revisions and updates, incorporating relevant data sources and current testing and maintenance intervals. The Bruce PRA Guide B-REP-03611-00005 R001 specifies that a realistic approach should be applied to probabilistic analysis, wherein realistic assumptions and data are used and unnecessary conservatism is avoided. Some conservatism may be acceptable where information is lacking or there is a high level of uncertainty, in order to avoid unjustifiable optimism, or where risk insights from sensitivity assessments indicate low impact on results. For determination of plant response and success criteria (both in event tree and fault tree analyses), especially for design basis accidents (DBAs), the existing BP PRAs rely on the conservative safety analyses that are described in the plant-specific Safety Reports. (The current version of the Bruce B Safety Report containing deterministic safety analysis is NK29-SR-01320-00002 R005.) However, when a structure, system or component (SSC) is identified as providing a specific	

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		mitigating function for a beyond design basis accident (BDBA), the conservative safety analysis should only be used if the assumptions in the safety analysis are not risk important. For risk important assumptions the supporting analysis should be made as realistic as possible. Examples of supporting analyses for PRA include the use of the MAAP-CANDU code in the Level 2 Bruce B PRA Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 (NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power letter, F. Saunders to R. Lojk, December 24, 2013) to provide best estimate analysis for determining accident progression and timing. Similarly, MAAP-CANDU has also been used in a consequence assessment for select internal events occurring during outage states defined in the Level 1 Outage PRA, to establish a technical basis that the consequences arising from initiating events during outage are bounded by those encountered at-power and documented in the Level 2 PRA. Bruce B PRAs have been supported by deterministic analysis, according to the systematic process defined in the Bruce PRA Guides. For example, the Level 1 At-Power Internal Events PRA Guide B-REP-03611-00005 R001 defines a systematic process for identifying initiating events for PSA, which calls for a review of the deterministic safety analysis: "The deterministic accident analyses should be reviewed to ensure that all relevant initiating events have been	

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		identified in the PRA. Sources of information include the plant-specific Safety Report, as well as other safety analysis documentation". The PRA guide B-REP-03611-00005 R001 also provides for the use of expert judgement to support the preparation of a PRA when there is a lack of information or analytical methods for resolving a specific technical issue. For example, engineering judgement is used in the Level 1 Internal Events PRA (B1401/RP/004 R01, July 18, 2014) where subjective failure probabilities are assigned to undeveloped events present in the fault trees. Another example where assessments are used to support the PRA model is the development of an improved inter-unit feedwater tie header model to replace subjective credits for the equipment survival under harsh environment with the information from powerhouse emergency venting system (PEVS) Margin Assessment as part of a PRA model change in 2011 (see Appendix F of the Bruce B Risk Assessment (BBRA) Level 1 At Power PRA Model Integration Report including Emergency Mitigation Equipment (EME), B1401/RP/004 R01, July 18, 2014).	
4.6	The level of detail of the PSA shall be consistent with the facility testing, maintenance and configuration management programs, and should be consistent with the intended uses of the PSA.	The current level of detail in BBRA is consistent with testing and configuration management for the Bruce B units. The current Bruce Power PRAs are updated approximately annually to capture changes to the plant as well as to operational practices (testing, maintenance, etc.). Bruce B PRAs are prepared under the general	C

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		process described in the BP governing document DIV-ENG-00010-R000, which establishes the requirements for the use of PRA at BP nuclear facilities. Within this framework, department procedure DPT-RS-00007 R001 provides instructions for the preparation and maintenance of plant-specific PRAs, defines the process for preparing a PRA as well as the systematic process of updating the PRA in order to maintain it as a "Living PRA". In particular, the regular updates of the BBRA model incorporate accumulated significant changes stemming from design, operational, maintenance, analysis and PRA applications experience, as required to keep the PRA consistent with the as built and as operated state of the plant. The process of periodic risk reassessment in PRA, as defined in the procedure DPT-RS-00007-R001, is based on changes from significant operational events, or changes to engineering, operations, surveillance and maintenance, evaluations of risk outside the scope of the existing PRA, design changes and component reliability updates (e.g., Bruce B Annual Reliability Report 2015, NK29-REP-09051.1-00016 R000 (see Enclosure 1 of NK29-CORR-00531-13197, Bruce B Annual Reliability Report – 2015, Bruce Power Letter, F. Saunders to K. Lafrenière, April 28, 2016)), issues from operating experience, etc. A full summary of updates of the BBRA model is given in Appendix F of the current Level 1 At Power Model Integration Report, B1401/RP/004 R01, July 18, 2014 (NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk	

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		Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014). Bruce Power intends to continue to maintain and update BBRA for consistency with NPP testing and configuration management, taking into account the currently pursued asset management initiative and associated with its risk-informed decision making practices.	
4.7	Seek CNSC acceptance of the methodology and computer codes to be used for the PSA before using them for the purposes of this document. Guidance: The methodology should be suitable to support the objectives of the PSA (set forth in section 3 of this document) and to support the intended PSA applications. Acceptance of the methodology prior to actual PSA development aims to help ensure the methodology can support the PSA's objectives. For example, the computer codes that support the analytical methods should be adequate for the purpose and scope of the analysis. Note: At the time of publication, the CNSC was reviewing the methodology for developing multi- unit PSA to evaluate the site integrated risk. The CNSC will establish the safety goals for site-wide PSA, which may consider:	The current BBRA models employ standard fault tree and event tree methodologies, implemented through the use of the Computer Assisted Fault Tree Analysis (CAFTA) software that is commonly used in nuclear industry for probabilistic risk modelling. The Modular Accident Analysis Program MAAP4-CANDU is used to perform consequence analysis for severe accidents. Other computer codes and utility programs are also used in PSAs. These codes have been submitted by Bruce Power to the CNSC for acceptance as per: • Bruce Power Probabilistic Risk Assessment Computer Codes, NK21-CORR-00531-10451/NK29-CORR-00531-10851, F. Saunders to R. Lojk, July 9, 2013, • Bruce Power Probabilistic Risk Assessment Additional Computer Code Information, NK21-CORR-00531-10630/NK29-CORR-00531-11021, F. Saunders to R. Lojk, August 14, 2013, • Bruce Power Probabilistic Risk Assessment Computer Codes, NK21-CORR-00531-	C

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	• interactions between the units, due to an initiating event (single-unit events and common-mode events), or as a result of the accident progression • aggregation of risk from internal events, internal hazards, and external hazards during all operating modes for all units at a site • radioactive sources other than the reactor cores (noting that alternate analysis methods may be used if accepted by the CNSC)	10744/NK29-CORR-00531-11128, F. Saunders to R. Lojk, October 11, 2013, • Bruce Power Probabilistic Risk Assessment - Additional Computer Code Information, NK21-CORR-00531-11043/NK29-CORR-00531-11439, F. Saunders to R. Lojk, February 24, 2014. The CNSC provided acceptance for use in Level 1 & 2 PRA of the computer codes CAFTA, SYSIMP, PRAQUANT, and RISKSPECTRUM as per: • Bruce A & B Risk Assessment Action Plan - Computer Codes Documentation, NK21-CORR-00531-08531/NK29-CORR-00531-09287, K. Lafrenière to F. Saunders, February 28, 2011. The CNSC have also provided acceptance on the balance of computer codes used by Bruce Power in Level 1 & 2 PRAs as per: • Acceptance of Bruce Power Probabilistic Risk Assessment Computer Codes, NK21-CORR-00531-10877/NK29-CORR-00531-11258, R. Lojk to F. Saunders, October 29, 2013, and • Acceptance of Bruce Power Probabilistic Risk Assessment Computer Code - WINDFAIL, NK21-CORR-00531-11206/NK29-CORR-00531-11609, K. Lafrenière to F. Saunders, March 19, 2014. As mentioned in the assessment to clause 4.2 provided above in this table, Bruce Power has developed specific PRA guides that are used to describe the technical details of the PRA methodology and serve as reference documents for PRA	

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		developers, practitioners and other knowledgeable stakeholders. These detailed guides have been submitted to the CNSC for acceptance as follows: • Level 1 At-Power Internal Events PRA Guide (B-REP-03611-00005-R001) • Level 1 Outage Internal Events PRA Guide (B-REP-03611-00006-R000) • Level 2 At-Power Internal Events PRA Guide (B-REP-03611-00010-R001) • Internal Flood PRA Guide (B-REP-03611-00007 R002, see also NK21-CORR-00531-10067/NK29-CORR-00531-10486) • Internal Fire PRA Guide (B-REP-03611-00008 R001, see also NK21-CORR-00531-10067/NK29-CORR-00531-10486) • Seismic PRA Guide (B-REP-03611-00009 R001, see also NK21-CORR-00531-09763/NK29-CORR-00531-10249) • External Hazards Screening and Disposition Guide (B-REP-03611-00011 R001) • High Wind PRA Guide (B-REP-03611-00012 Rev 0) • Common Cause Failure Methodology for Bruce Power PRA (B0978/RP/002 R001, November 21, 2011, see also NK21-CORR-00531-09019/NK29-CORR-00531-09699) CNSC's acceptance of the aforementioned guides has been provided via the following CNSC	

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		correspondence: • NK21-CORR-00531-08908/NK29-CORR-00531-09623, August 31, 2011 – for the Level 1 At-Power and Outage guides • NK21-CORR-00531-10191/NK29-CORR-00531-10595, January 21, 2013 – for the Level 2 At-Power PRA guide • NK21-CORR-00531-10212/NK29-CORR-00531-10614, January 28, 2013 – for the Internal Flood PRA guide • NK21-CORR-00531-10193/NK29-CORR-00531-10599, January 21, 2013 – for the Internal Fire PRA guide • NK21-CORR-00531-10638/NK29-CORR-00531-11030, July 8, 2013 – for the Seismic PRA guide • NK21-CORR-00531-10263/NK29-CORR-00531-10672, February 20, 2013 – for the High Wind PRA guide • NK21-CORR-00531-10364/NK29-CORR-00531-10776, March 25, 2013 – for the CCF Methodology Development of a whole-site PSA methodology for CANDU reactors is the subject of COG-JP-4499-001-R0 (Project Execution Plan to Implement COG Whole-site PSA Methodology, Kinectrics Report No: K-410085-REPT-0001, Rev 01). Although Bruce Power considers simple aggregation of risk from internal events, internal hazards and	

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		external hazards as an inaccurate method to determine risk due to the potential for double counting in some areas, results of a simple addition of SCDFs and LRFs from the existing PRAs for internal and external events was reported in the July 2014 submission from BP to CNSC (see Section 2.0 of Attachment B to NK21-CORR-00531-11324/NK29-CORR-00531-11729, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014). This estimate demonstrates that, when the PRA model credits recent and ongoing plant improvement (such as the installation of the Fukushima-related EME, automatic isolation of the Shield Tank Expansion line and containment enhancements), the risk aggregation results meet the Bruce Power single unit SCDF and LRF limits of 1.0E-4/yr and 1.0E-5/yr, respectively, as provided in the Level 2 At-Power Internal Events PRA Guide (B-REP-03611-00010 R001). Safety assessment of the irradiated fuel bay (which is a source of radiation other than the reactor core) has been conducted outside the scope of PRA, as documented in NK21-CORR-00531-10341/NK29-CORR-00531-10750, Bruce Power Irradiated Fuel Bay Structural Integrity Analysis, Bruce Power Letter, F. Saunders to R. Lojk, March 26, 2013. This analysis was reviewed by CNSC and found acceptable per NK21-CORR-00531-10565/NK29-CORR-00531-10965, CNSC Review of Bruce Power's Irradiated Fuel Bay Structural Integrity Analysis (Fukushima Action Items 1.5.1, 1.6.1 and 1.6.2), CNSC Letter, R. Lojk to F. Saunders, June 3, 2013).	

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4.8	Include all potential site-specific initiating events and potential hazards, namely: • internal initiating events and internal hazards • external hazards, both natural and human-induced, but non-malevolent Include potential combinations of the external hazards. The screening criteria of hazards shall be acceptable to the CNSC. The licensee may, with the agreement of "persons authorized" by the Commission, choose an alternate analysis method to conduct the assessment of internal hazards and external hazards. Guidance: Examples of external hazards are seismic hazards, external fires (e.g. fires affecting the site and originating from nearby forest fires), external floods, high winds, off-site transportation accidents, releases of toxic substances from off-site storage facilities, severe weather conditions. Examples of internal hazards are internal fires, internal floods, turbine missiles, onsite transportation accidents, and releases of toxic substances from onsite storage facilities.	The Initiating Events (IEs) included in the PRAs are plant-specific. Their selection and quantification is based on the CNSC accepted procedures described in the Level 1 PRA guides for the at-power and outage states, B-REP-03611-00005 R001 and B-REP-03611-00006 R000, respectively, and in the Level 2 Internal Events PRA Guide B-REP-03611-00010 R001. As previously mentioned in the assessment for clause 4.1 above, the latest PRAs for Level 1 internal events at-power and outage, as well as for Level 2 internal events have been submitted to the CNSC as follows: 1) 2014 Bruce B Risk Assessment (BBRA) Level 1 At-Power Model Integration Report including Emergency Mitigation Equipment (EME)), B1401/RP/004 R01, July 18, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] 2) 2014 Bruce B Risk Assessment (BBRA) Level 1 Outage PRA Model Summary Report, B1401/RP/002 R01, July 17, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] 3) 2013 Bruce B PRA Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 [NK21-CORR-00531-10958/NK29-CORR-00531-11342,	C

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		Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] The following latest revisions of the Bruce B PRAs for internal hazards include: 4) At-Power Internal Fire PRA: K-410003-REPT-0037 R001 (see Enclosure 6 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of 5-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014) 5) At-Power Internal Flood PRA: K-410009-REPT-0011 R000 (see Enclosure 13 of NK21-CORR-00531-10958/NK29-CORR-00531-11342, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013) The following latest revisions of the Bruce B PRAs for external hazards include: 6) At-Power Seismic PRA: K-410003-REPT-0074 R001 (see Enclosure 4 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014 7) At-Power High Wind PRA: B1401/RP/006 R001 (see Enclosure 9 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports,	

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		Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014) For external hazards other than seismic and high wind, external hazards assessments have been performed. A list of relevant external hazards related reports, submissions and correspondence is provided below in chronological order. 8) External Hazards Assessment Phase 1, K-449958-0007 R001, August 29, 2012, was submitted to the CNSC via NK21-CORR-00531-09809/NK29-CORR-00531-10287, Bruce A and B External Hazards Assessment, Bruce Power Letter, F. Saunders to R. Lojk, September 28, 2012. 9) CNSC provided comments on the above submission via NK21-CORR-00531-10637/NK29-CORR-00531-11029, Bruce A and B External Hazard Assessment At Power, CNSC Letter, R. Lojk to F. Saunders, July 8, 2013. 10) CNSC accepted Bruce Powers External Hazards Assessment Phase 1 via NK21-CORR-00531-10972/NK29-CORR-00531-11360, CNSC Assessment of Bruce A and B External Hazard Assessment At-Power, CNSC Letter, R. Lojk to F. Saunders, December 9, 2013. 11) Bruce Power submitted additional assessments of external hazards as Phase 2a via NK21-CORR-00531-10848/NK29-CORR-00531-11226, Bruce A and B External Hazard Assessment, Bruce Power Letter, F. Saunders to R. Lojk, December 12, 2013.	

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		12) Bruce Power submitted report K-449958-0014 R00 0 via Enclosure 14 of NK21-CORR-00531-10958/NK29-CORR-00531-11342, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013, summarizing all Phase 1 and Phase 2 assessments performed to-date. 13) Bruce Power submitted report K-449958-0016 R001 via Enclosure 7 of NK21 -CORR-00531-11091/NK29-CORR-00531-11491, S-294 Probabilistic Risk Assessment Confirmatory Activities, Bruce Power Letter F. Saunders to K. Lafrenière, March 31, 2014. This provided an update for the assessments done to-date and summarized the expectation for the outstanding External Flooding assessment. 14) Bruce Power submitted the External Flooding Assessment report K-449958-0007 R003 via Enclosure 7 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014. Multiple combinations of the external hazards were considered in the Phase 1 and Phase 2 hazards assessments; the justification of screening these combinations out are documented in Appendix A of the aforementioned report (K-449958-REPT-0007 R003). 15) A limited PRA Level 2 outage study has been performed and submitted to CNSC as Enclosure 1 of	

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		the January 22, 2014 Bruce Power submission to CNSC NK21-CORR-00531-11017/NK29-CORR-00531-11413, Outage Probabilistic Risk Assessments (PRA) for Internal Fires, Seismic Events and Internal Floods, F. Saunders to R. Lojk (see Bruce NGS A and B Level 2 Outage Probabilistic Risk Assessment, B1380/RP/001 R001, September 13, 2013). 16) Note also that the January 22, 2014 submission to CNSC NK21-CORR-00531-11017/NK29-CORR-00531-11413, Outage Probabilistic Risk Assessments (PRA) for Internal Fires, Seismic Events and Internal Floods, F. Saunders to R. Lojk, provides justification why outage PRAs for internal fires, seismic events and internal floods do not need to be performed for S-294 compliance (the justification is also valid with respect to the REGDOC-2.4.2 compliance). 17) In NK21-CORR-00531-11284/NK29-CORR-00531-11692, Outage Probabilistic Risk Assessment (PRA) for Internal Fires, Seismic Events and Internal Floods, K. Lafrenière to F. Saunders, April 29, 2014, the CNSC has accepted Bruce Power's arguments provided in correspondence NK21-CORR-00531-10364//NK29-CORR-00531-11413 for not needing to perform outage PRAs for internal fires, seismic events and internal floods.	
4.9	Include at-power and shutdown states. A PSA shall also be performed for other states where the reactor is expected to operate for extended periods of time and that are not covered by the at-power and shutdown PSAs.	Bruce B PRAs cover both at-power and shutdown (outage) states. Most recent examples of at-power PRAs and relevant documentation are as follows: 1. 2013 Bruce B PRA Level 1 At-Power Summary Report, B1294/RP/002 R01, August 2013 [NK21-	C

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		CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] 2. 2013 Bruce B PRA Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 [NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] 3. 2014 Bruce B Risk Assessment (BBRA) Level 1 At-Power Model Integration Report including Emergency Mitigation Equipment (EME), B1401/RP/004 R01, July 18, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 11, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014] 4. At-Power Internal Fire PRA: K-410003-REPT-0037 R001 (see Enclosure 6 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014) 5. At-Power Internal Flood PRA: K-410009-REPT-0011 R000 (see Enclosure 13 of NK21-CORR-00531-10958/NK29-CORR-00531-11342, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013)	

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		6. At-Power Seismic PRA: K-410003-REPT-0074-R001 (see Enclosure 4 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014) 7. At-Power High Wind PRA: B1401/RP/006 R001 (see Enclosure 9 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014) Most recent examples of shutdown PRAs and relevant documentation are as follows: 8. 2014 Bruce B Risk Assessment (BBRA) Level 1 Outage PRA Model Summary Report, B1401/RP/002 R01, July 17, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014] 9. Per Attachment A of correspondence [NK21-CORR-00531-11091/NK29-CORR-00531-11491, S-294 Probabilistic Risk Assessment Confirmatory Activities, Bruce Power letter, F. Saunders to K. Lafrenière, March 31, 2014], Bruce Power concludes that there is no need to complete a Level 2 outage PRA on the basis of the following: • Level 1 Outage PRA results characterize the risks encountered during outages. The results satisfy	

Article No.	Clause Requirement	Assessment	Compliance Category
		the CDF safety goal with good margin; • Level 2 At-Power PRA results and insights provide significant information on the behaviour of containment during a core damage event and effectively bound the contribution of outage unit accidents to release scenarios; • MAAP analysis undertaken specifically for the outage state show that only Drained Guaranteed Shutdown State sequences have the potential for large release. If the CDF of this state is conservatively assumed to lead directly to LRF, the safety goal is still met. The result of this bounding assessment obviates the need for further outage Level 2 considerations. 10. Note also that the January 22, 2014 submission to CNSC NK21-CORR-00531-11017/NK29-CORR-00531-11413, Outage Probabilistic Risk Assessments (PRA) for Internal Fires, Seismic Events and Internal Floods, F. Saunders to R. Lojk, provides justification why outage PRAs for internal fires, seismic events and internal floods do not need to be performed for S-294 compliance (the justification is also valid with respect to the REGDOC-2.4.2 compliance). 11. In NK21-CORR-00531-10364//NK29-CORR-00531-11692 of April 29, 2014, the CNSC has accepted Bruce Power’s arguments provided in correspondence NK29-CORR-00531-11413 for not needing to perform outage PRAs for internal fires, seismic events and internal floods.	

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		12. Within the scope of the limited PRA Level 2 outage study (see Bruce NGS A and B Level 2 Outage Probabilistic Risk Assessment, B1380/RP/001 R001, September 13, 2013, submitted to the CNSC as Enclosure 1 of the January 22, 2014, correspondence NK21-CORR-00531-11017/NK29-CORR-00531-11413), there exists a limited consequence assessment for a Unit Loss-of-Heat Sinks (ULHS) event and for a Small Loss-of-Coolant Accident (SLOCA) caused by ice plug failure on a low-elevation feeder. It is also noteworthy that the at-power and outage PSAs bound the PSA results for other states.	
4.10	Include sensitivity analysis, uncertainty analysis and importance measures in the PSA.	Sensitivity analysis provides information regarding the change in the important outputs of a deterministic model with respect to changes in the model inputs considering perturbation of one input at a time. Of particular interest is to identify those parameters for which a small change can result in a major impact on analysis outcomes. In addition, the results of sensitivity analysis provide valuable input to uncertainty analysis, by ensuring that the uncertainty in parameters to which the key results are sensitive are captured in the uncertainty analysis. For Level 1 PRA at-power and outage PRAs, the sensitivity effects on SCDF are sought, while for Level 2 PRAs the sensitivity effects on LRF are sought, as the SRF value is well below the Bruce Power's safety limit of 1.0E-04/r-yr (as provided in the Level 2 At-Power Internal Events PRA Guide B-REP-03611-00010	C

Article No.	Clause Requirement	Assessment	Compliance Category
		R001). Within the Level 1 At-Power PRA, the investigated model methodologies and assumptions included modelling of Common Cause Failures (CCFs), Non-Occurrence of Gland Seal LOCA, Impact of ECI Header Pressurization, Impact of Crediting Emergency Water System (EWS) as Source of HTS Make-up, and Impact of EWS Failure Criteria. For details see: 1) 2013 Bruce B PRA Level 1 At-Power Summary Report, B1294/RP/002 R01, August 2013 [NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] Within the Level 1 Outage PRA, sensitivity analyses included parameter changes describing Event Tree Human interaction (HI) dependencies, Moderator firewater addition, Common Cause Failures events, EWS Pump Failure Criterion, etc. For details see: 2) 2014 Bruce B Risk Assessment (BBRA) Level 1 Outage PRA Model Summary Report, B1401/RP/002 R01, July 17, 2014 [NK21-CORR-00531-11324/NK29-CORR-00531-11729, Enclosure 2, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014] The sensitivity analyses performed within the Level 2 At-Power PRA include two kinds of studies: effects of Level 1 and Level 2 parameter changes on LRF and	

Article No.	Clause Requirement	Assessment	Compliance Category
		effects of parameter changes on deterministic consequence modelling done by MAAP-CANDU. For details see: 3) 2013 Bruce B PRA Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 [NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] In Level 1 BBRA PRA, a measure of the uncertainty associated with the BBRA SCDF was obtained by means of UNCERT utility code. For this study, the Monte Carlo method was used to propagate basic event uncertainty and develop the resulting uncertainty distribution for the severe core damage frequency. For details see both reference items 1 and 2 mentioned above (for both at-power and outage uncertainly analysis). In Level 2 PRA, a Monte Carlo method was used to estimate the distribution associated with each release category (RC) frequency, as well as with LRF and SRF. This method randomly selects a value in the uncertainty distribution assigned to the values of parameters for calculating the probability of the event sequences that contribute to the RC, LRF and SRF results. In the context of a Level 2 PRA the sources of uncertainty are those introduced by the Level 2 analysis (including phenomenological modeling uncertainties and containment event tree uncertainties) in addition to the uncertainties introduced by the inputs derived from Level 1 PRA.	

Article No.	Clause Requirement	Assessment	Compliance Category
		For details see reference item 3 mentioned above. Sensitivity and uncertainty are taken into account in the Seismic PRA. Consideration of uncertainty is provided by evaluation of parametric as well as of sources of model uncertainties. A number of sensitivity evaluations were performed to consider the sensitivity of the results to various modeling assumptions, such as: hazard curve binning, human action, PHT leak, etc. For details see: 4) At-Power Seismic PRA: K-410003-REPT-0074 R001 (see Enclosure 4 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014 Sensitivity and uncertainty are taken into account also in the Fire PRA. For details see section 3.15 of: 5) At-Power Internal Fire PRA: K-410003-REPT-0037 R001 (see Enclosure 6 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of 5-294 Probabilistic Risk Assessment Final Reports, Bruce Power letter, F. Saunders to K. Lafrenière, July 31, 2014) Importance measures are used in PRA models to establish the significance of the events/systems in the fault trees in terms of their quantitative contribution to the SCDF, SRF and LRF frequencies. The methodology applied by Bruce Power is consistent with the best industry practices, i.e., evaluating importance of systems, components and HI events to	

Article No.	Clause Requirement	Assessment	Compliance Category
		SCDF, SRF and LRF using Fussell-Vesely (FV) and Risk Achievement Worth (RAW) importance measures and evaluating importance of initiating events using the FV measure. For details see reference items 1 to 3 mentioned above (for Level 1 at-power and outage, and Level 2 at-power importance analysis).	

B.2. 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 REGDOC-2.5.2 have been assessed in Table B2. A more detailed assessment is performed in “Safety Factor 1 – Plant Design”.

Table B2: CNSC REGDOC-2.5.2, Design of Reactor Facilities: Nuclear Power Plants

Article No.	Clause Requirement	Assessment	Compliance Category
4.2.2	*Qualitative safety goals* A limit is placed on the societal risks posed by NPP operation. For this purpose, the following two qualitative safety goals have been established: Individual members of the public shall be provided a level of protection from the consequences of NPP operation, such that there is no significant additional risk to the life and health of individuals. Societal risks to life and health from NPP operation shall be comparable to or less than the risks of generating electricity by viable competing technologies, and shall not significantly add to other societal risks. *Quantitative application of the safety goals* For practical application, quantitative safety goals have been established, so as to achieve the intent of the qualitative safety goals. The three quantitative safety goals are: 1. core damage frequency	The quantitative safety goals calculated in the Bruce B Probabilistic Risk Assessment (PRA) are defined in accordance with the requirement of this clause. However, the limiting values of the safety goals adopted in the Bruce B PRA are one order of magnitude larger than the corresponding limits required in the clause, i.e. Bruce B PRA uses the safety goal limits defined in the Level 2 PRA Guide B-REP-03611-00010 R001: • for the severe core damage frequency to be less than 1E-4 per reactor year; • for the small release frequency to be less than 1E-4 per reactor year; • for the large release frequency to be less than 1E-5 per reactor year. The following results of the Bruce B PRAs are summarized in the letter NK21-CORR-00531-11324/NK29-CORR-00531-11729, submitted to the CNSC on July 31, 2014, and in B1538/005/000001:	Gap

Article No.	Clause Requirement	Assessment	Compliance Category
	2. small release frequency 3. large release frequency A core damage accident results from a postulated initiating event (PIE) followed by the failure of one or more safety system(s) or safety support system(s). Core damage frequency is a measure of the plant's accident prevention capabilities. Small release frequency and large release frequency are measures of the plant's accident mitigation capabilities. They also represent measures of risk to society and to the environment due to the operation of an NPP. *Core damage frequency* The sum of frequencies of all event sequences that can lead to significant core degradation shall be less than 10^{-5} per reactor year. *Small release frequency* The sum of frequencies of all event sequences that can lead to a release to the environment of more than 10^{15} becquerels of iodine-131 shall be less than 10^{-5} per reactor year. A greater release may require temporary evacuation of the local population. *Large release frequency* The sum of frequencies of all event sequences that	Severe Core Damage Frequency (SCDF) for At-Power Internal Events: 5.18E-6 per reactor year (if Emergency Mitigating Equipment (EME) installed for Fukushima-related improvements are credited), or 1.48E-5 per reactor year (without crediting the Fukushima-related EME as obtained in the Level 1 At-Power Internal Events, B1294/RP/002 R01, August 2013 [see Enclosure 2 to NK21-CORR-00531-10958/NK29-CORR-00531-11342, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013)] SCDF for Outage Internal Events: 8.30E-6 per reactor year SCDF for Internal Flood: 4.60E-7 per reactor year (with the Fukushima-related EME credited) SCDF for Fire Hazard: 4.06E-6 per reactor year (with the Fukushima-related EME credited)	

Article No.	Clause Requirement	Assessment	Compliance Category
	can lead to a release to the environment of more than 10^{14} becquerels of cesium-137 shall be less than 10^{-6} per reactor year. A greater release may require long term relocation of the local population Guidance A comprehensive probabilistic safety assessment (PSA) considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates for the safety of the plant. Core damage frequency is determined by a Level 1 PSA, which identifies and quantifies the sequence of events that may lead to significant core degradation. The small release frequency and large release frequency are determined by a Level 2 PSA, which starts from the results of a Level 1 PSA, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment. An exemption for performing a Level 2 PSA is granted if it is shown that core damage frequency in the Level 1 PSA is sufficiency low (i.e., less than the large release frequency limit). Calculations of the safety goals include all internal and external events as per REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. However, aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release	SCDF for Seismic Hazard: 7.20E-7 per reactor year (crediting the Fukushima-related EME) SCDF for High Wind Hazard: 6.16E-6 per reactor year (crediting the Fukushima-related EME) *Aggregated SCDF* obtained by summation of the above SCDFs: 2.49E-5 per reactor year (with the Fukushima-related EME credited) Large Release Frequency (LRF) for At-Power Internal Events: 6.93E-7 per reactor year (if Emergency Mitigating Equipment (EME) installed for Fukushima-related improvements are credited, as reported in the document “RE: Bruce A and Bruce B Level 2 At-Power PRA Results Including Emergency Mitigating Equipment” B1538/005/000001, November 20, 2014), or 5.49E-6 per reactor year (without crediting the Fukushima-related EME, as	

Article No.	Clause Requirement	Assessment	Compliance Category
	frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated. Further details on PSAs are contained in section 9.5 of this document and REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants.	obtained in the Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 [see NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] LRF for Fire Hazard: 8.74E-7 per reactor year (with the Fukushima-related EME credited) LRF for Seismic Hazard: 7.20E-7 per reactor year (with the Fukushima-related EME credited) LRF for High Wind Hazard: 6.16E-6 per reactor year (crediting the Fukushima-related EME) *Aggregated LRF* obtained by summation of the above LRFs: 8.45E-6 per reactor year (with the Fukushima-related EME credited) *Small Release Frequency* (SRF) for At-Power Internal	

Article No.	Clause Requirement	Assessment	Compliance Category
		Events: 7.14E-7 per reactor year (if Emergency Mitigating Equipment (EME) installed for Fukushima-related improvements are credited, as reported in the document “RE: Bruce A and Bruce B Level 2 At-Power PRA Results Including Emergency Mitigating Equipment” B1538/005/000001, November 20, 2014), or 5.67E-6 per reactor year (without crediting the Fukushima-related EME, as obtained in the Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 [see NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013] If Fukushima-related EMEs are credited, all SCDFs for individual events meet both the Bruce Power guide’s (B-REP-03611-00010 R001) safety goal limit of 1E-4 per reactor year and CNSC’s REGDOC-2.5.2 clause 4.2.2 safety goal limit of 1E-5 per reactor year. The SRF for at-power internal events meets both the Bruce Power guide’s (B-REP-03611-00010 R001) safety goal limit of 1E-4 per reactor year and CNSC’s REGDOC-2.5.2 clause 4.2.2 safety goal limit of 1E-5 per reactor year. If Fukushima-related EMEs are credited, all LRFs but	

Article No.	Clause Requirement	Assessment	Compliance Category
		for one of the events meet both the Bruce Power guide's (B-REP-03611-00010 R001) safety goal limit of 1E-5 per reactor year and CNSC's REGDOC-2.5.2 clause 4.2.2 safety goal limit of 1E-6 per reactor year. The PRA for high wind events results in an LRF of 6.16E-6, which is higher than the REGDOC's limit of 1E-6 per reactor year. However, the following has to be noted: • The SCDF for the Bruce B High Wind PRA model is 6.16E-6 occurrences/year. The LRF is bounded by SCDF and thus is less than 6.16E-6 occurrences/year, although it may still be higher than 1E-6 per year; • Moreover, the high wind PRA is done for the site rather than for individual units. A refinement to this approach is to review and identify single unit cutsets for high wind events. For these cutsets, additional failures would have to occur in order to result in severe accidents leading to a large release. Given that the SCDF values are within the same range with the limit, it is expected that adopting this approach would enable Bruce B High Wind PRA results to meet the LRF REGDOC limit. Although the result of each individual PRA meets the safety goal limits set up for Bruce B PRAs (with the exception of high wind LRF result as noted above), their aggregates obtained by respective summation of SCDFs and LRFs do not meet the more stringent quantitative safety goal targets set up in the REGDOC requirement clause. Therefore, a gap is assessed	

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		against this clause.	
4.2.3	To demonstrate achievement of the safety objectives, a comprehensive hazard analysis, a deterministic safety analysis, and a probabilistic safety assessment shall be carried out. These analyses shall identify all sources of exposure, in order to evaluate potential radiation doses to workers at the plant and to the public, and to evaluate potential effects on the environment. The safety analyses shall examine plant performance for: 1. normal operation 2. AOOs 3. DBAs 4. BDBAs, including DECs (DECs could include severe accident conditions) Based on these analyses, the capability of the design to withstand PIEs and accidents shall be confirmed, the effectiveness of the items important to safety demonstrated, and requirements for emergency response established. The results of the safety analyses shall be fed back into the design. The safety analyses are discussed in further detail in section 9.0.	The part of this clause regarding probabilistic safety assessment is covered in detail in the assessment of CNSC REGDOC-2.4.2	RNA

Article No.	Clause Requirement	Assessment	Compliance Category
5.6	Safety assessment is a systematic process applied throughout the design phase to ensure that the design meets all relevant safety requirements. The safety assessment for the design shall include the requirements set by the operating organization and by regulatory authorities. The basis for the safety assessment shall be the data derived from the safety analysis, previous operational experience, results of supporting research, and proven engineering practices. The safety assessment shall be part of the design process, with iteration between the design and analyses, and shall increase in scope and level of detail as the design process progresses. Before the design is submitted, an independent peer review of the safety assessment shall be conducted by individuals or groups separate from those carrying out the design. Safety assessment documentation shall identify those aspects of operation, maintenance and management that are important to safety. This documentation shall be maintained in a dynamic suite of documents, to reflect changes in design as the plant evolves. Safety assessment documentation shall be presented clearly and concisely, in a logical and understandable format, and shall be made readily accessible to designers, operators and the CNSC.	The part of this clause regarding probabilistic safety assessment is covered in detail in the assessment of CNSC REGDOC-2.4.2.	RNA

Article No.	Clause Requirement	Assessment	Compliance Category
	Guidance As per IAEA GSR Part 4, Safety Assessment for Facilities and Activities, aspects considered in the safety assessment should include: • defence in depth • safety margins • multiple barriers • safety analysis (including both deterministic and probabilistic approaches), as well as overall scope, approach, safety criteria, uncertainty and sensitivity analysis, use of computer codes, and use of operating experience • radiation risks • safety functions • site characteristics • radiation protection • engineering aspects • human factors • long-term safety The independent peer review should be performed by suitably qualified and experienced individuals. Additional information Additional information may be found in: • IAEA, GSR Part 4, *Safety Assessment for Facilities and Activities*, Vienna, 2009.		

Article No.	Clause Requirement	Assessment	Compliance Category
7.4	The design for the NPP shall apply a systematic approach to identifying a comprehensive set of postulated initiating events, such that all foreseeable events with the potential for serious consequences or with a significant frequency of occurrence are anticipated and considered. Postulated initiating events can lead to AOOs, DBAs or BDBAs, and include credible failures or malfunctions of SSCs, as well as operator errors, common-cause internal hazards, and external hazards. For a site with multiple units, the design shall take due account of the potential for specific hazards simultaneously impacting several units on the site. Guidance The postulated initiating events (PIEs) are identified using engineering judgment and deterministic and probabilistic assessment. A justification of the extent of usage of deterministic safety analyses and probabilistic safety analyses should be provided, in order to show that all foreseeable events have been considered. Sufficient information should be provided regarding the methods used to identify PIEs, their scope and classification. In cases where the identification methods have made use of analytical tools (e.g., master logic diagrams, hazard and operability analysis, failure modes and effect analysis), detailed	The part of this clause regarding probabilistic safety assessment is covered in detail in the assessment of CNSC REGDOC-2.4.2.	RNA

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	information is expected to be presented. A systematic approach to event classification should consider all internal and external events, all normal operating configurations, various plant and site conditions, and failure in other plant systems (e.g., storage for irradiated fuel, and tanks for radioactive substances). The design should take into account failure of equipment that is not part of the NPP, if the failure has a significant impact on nuclear safety. CNSC REGDOC-2.4.1, Deterministic Safety Analysis and CNSC REGDOC-2.4.2, Probabilistic Safety Assessments, provide the requirements and guidance for establishing the scope of PIEs, and for classifying the PIEs in accordance with their anticipated frequencies, and other factors, as appropriate. For further information on the safety analysis for the identified PIEs, refer to section 9.0 of this document. Additional information Additional information may be found in: • CNSC, REGDOC-2.4.1, Deterministic Safety Analysis, Ottawa, Canada, 2014.		
7.6	All SSCs important to safety shall be designed with sufficient quality and reliability to meet the design limits. A reliability analysis shall be performed for	Bruce B uses the reliability program described in BP-PROG-11.01 and in the hierarchy of its implementing procedures (listed in Appendix B of BP-PROG-11.01). The implementing procedures deal with scoping and	C

Article No.	Clause Requirement	Assessment	Compliance Category
	each of these SSCs. Where possible, the design shall provide for testing to demonstrate that the reliability requirements will be met during operation. The safety systems and their support systems shall be designed to ensure that the probability of a safety system failure on demand from all causes is lower than 10^{-3}. The reliability model for each system may use realistic failure criteria and best-estimate failure rates, considering the anticipated demand on the system from PIEs. Design for reliability shall take account of mission times for SSCs important to safety. The design shall take into account the availability of offsite services upon which the safety of the plant and protection of the public may depend, such as the electricity supply and external emergency response services. Guidance The design for reliability is based on meeting applicable regulatory requirements and industry standards. The design should provide assurance that the requirements of CNSC RD/GD-98, Reliability Programs for Nuclear Power Plants, will be met during operation. Not all SSCs important to safety identified in the design phase will necessarily be included in the	identification of critical Structure, System and Component (SSCs), continuing equipment reliability improvement, preventive maintenance implementation, performance monitoring, equipment reliability problem identification and resolution, long-term planning and life-cycle management. The decision methodology described in DPT-RS-00012 R0001 determines which plant systems meet the criteria of “Systems Important to Safety” (SIS). This identification incorporates the use of probabilistic unavailability models of SIS. The ongoing record of reliability of SIS is documented in Annual Reliability Reports. The 2015 Annual Reliability Report NK29-REP-09051.1-00016 (see Enclosure 1 of NK29-CORR-00531-13197, Bruce B Annual Reliability Report - 2015, Bruce Power Letter, F. Saunders to K. Lafrenière, April 28, 2016) contains detailed results on the Bruce B systems that comprise the SIS list. Quantitative unavailability models exist for nine of these systems; for others, CANDU Owner's Group (COG) guidance COG-05-9011 is followed, where the applicable initiating events frequencies are used as system monitoring parameters. Per the Bruce B 2015 Annual Reliability Report, all safety systems meet the requirement for the probability of failure on demand from all causes be lower than 1E-3. As per guidance provided by CNSC RD/GD-98, the resulting unavailabilities are assessed against their respective targets. The unavailability targets for the SIS were set out based on their design and	

Article No.	Clause Requirement	Assessment	Compliance Category
	reliability program. The following principles are applied for SSCs important to safety: • the plant is designed, constructed, and operated in a manner that is consistent with the assumptions and risk importance of these SSCs • these SSCs do not degrade to an unacceptable level during plant operations • the frequency of transients posing challenges to SSCs is minimized • these SSCs function reliably when challenged The reliability of SSCs assumed in the design stage needs to be realistic and achievable. Deterministic analysis or other methods may be used if the PSA lacks effective models or data to evaluate the reliability of SSCs.	operational requirements, per Section 2.3.2 of the COG guidance document COG-05-9011. In 2015, none of the twelve Systems Important to Safety exceeded their Bruce Power Predicted Future Unavailability (PFU) targets. Also per 2015 Annual Reliability Report (see Enclosure 1 of NK29-CORR-00531-13197, Bruce B Annual Reliability Report - 2015, Bruce Power Letter, F. Saunders to K. Lafrenière, April 28, 2016), Actual Past Unavailability (APU) was observed for four out of twelve Systems Important to Safety. The four systems were Emergency Coolant Injection System, Emergency Water System, Shutdown System One and Shutdown System Two. The APU for the Emergency Water System was above its target. Events that caused the high APU have been addressed through Bruce Power's corrective action process. The report also mentioned that there were five missed and twenty-two deferred Safety System Tests, and zero missed and thirteen deferred Predefined Maintenance items; the deferrals were evaluated using the BBRA and found to be acceptable based on system configuration and unavailability targets. Station Condition Records have been written to capture and trend these items.	
7.6.1	The potential for common-cause failures (CCFs) of items important to safety shall be considered in determining where to apply the principles of separation, diversity and independence so as to achieve the necessary reliability. Such failures could	The part of this clause regarding probabilistic safety assessment is covered in detail in the assessment of CNSC REGDOC-2.4.2.	RNA

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	simultaneously affect a number of different items important to safety. The event or cause could be a design deficiency, a manufacturing deficiency, an operating or maintenance error, a natural phenomenon, a human-induced event, or an unintended cascading effect from any other operation or failure within the plant. Guidance Failure of a number of devices or components to perform their functions could occur as a result of a single specific event or cause. CCFs could also occur when multiple components of the same type fail at the same time. This could be caused by occurrences such as a change in ambient conditions, saturation of signals, repeated maintenance error or design deficiency. Additional information Additional information may be found in: • United States Nuclear Regulatory Commission (U.S. NRC), NUREG/CR-7007, *Diversity Strategies for Nuclear Power Plant Instrumentation and Control Systems*, Washington, D.C., 2010. • U.S. NRC, Branch Technical Position (BTP) 7-19, *Guidance for Evaluation of Diversity and Defense-in-Depth and in Digital Computer-Based Instrumentation and Control Systems*,		

Article No.	Clause Requirement	Assessment	Compliance Category
	Washington, D.C., 2007. • U.S. NRC, NUREG/CR-6303, *Method for Performing Diversity and Defense-in-Depth Analyses of Reactor Protection Systems*, Washington, D.C., 1994.		
8.4.2	The design shall permit ongoing demonstration that each means of shutdown is being operated and maintained in a manner that ensures continued adherence to reliability and effectiveness requirements. Periodic testing of the systems and their components shall be scheduled at a frequency commensurate with applicable requirements. Guidance The reliability calculation should include sensing the need for shutdown, initiation of shutdown, and insertion of negative reactivity. All elements necessary to complete the shutdown function should be included. The reliability of the shutdown function should be such that the cumulative frequency of failure to shutdown on demand is less than 10^{-5} failures per demand, and the contribution of all sequences involving failure to shutdown to the large release frequency of the safety goals is less than 10^{-7}/yr. This considers the likelihood of the initiating event and recognizes that the two shutdown means may not be completely independent.	Only the Guidance portion of this clause is relevant to probabilistic safety analysis assessment (SF6). The Level 1 PRA At-Power Model Integration Report including EME B1401/RP/004 R01 (see also Enclosure 11 of NK21-CORR-00531-11324/NK29-CORR-00531-11729, Submission of S-294 Probabilistic Risk Assessment Final Reports, Bruce Power Letter, F. Saunders to K. Lafrenière, July 31, 2014) incorporates all sequences including failure to shutdown into the fuel damage category FDC1, whose value is estimated as 2.87E-8 occurrences per reactor per year. Thus the guidance target of cumulative frequency of failure to shut down on demand being less than 10^{-5}/yr is demonstrated by the fuel damage category FDC1 in the Level 1 PSA. The following information is extracted from Level 2 At-Power Summary Report, B0900/RP/055 R01, December 2013 (see NK21-CORR-00531-10958/NK29-CORR-00531-11342, Enclosure 4, Submission of S-294 Probabilistic Risk Assessment Deliverables, Bruce Power Letter, F. Saunders to R. Lojk, December 24, 2013): from Level 1 PRA, Fuel Damage Category 1 (FDC1) represents all sequences involving rapid accident progression resulting from failures to shutdown the reactor when required.	C

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	Section 7.6.2 requires that the shutdown function be delivered even in the presence of any single failure and even during the worst configuration from testing and maintenance. For example, for a rod based system to meet the SFC, the safety analysis may assume that the two highest worth control rods are unavailable (one for testing, and one assumed to fail on demand, in accordance with the SFC). In this case, no further testing of rods would be allowed until the rod under testing becomes available.	FDC1 is conservatively assumed to cause early consequential containment failure with a 0.5 probability and the failure sequence is assigned to a unique PDS, PDS1. Release Categories (RCs) are defined to bin the consequences associated with containment event tree end-states to facilitate comparison with safety goals. RC0 consists of single unit events (PDS1), two-unit events (PDS3A) and three- or four-unit events (PDS3). The contributions to RC0 of PDS3 and PDS3A are 94% and 4%, respectively, meaning that the contribution of PDS1 to RC0 is approximately 2%. The frequency of RC0 is included in the LRF calculation. RC0 frequency is 4.71E-6, which means that the contribution to it from PDS1 is 9.42E-8. This is below the target for the contribution of all sequences involving failure to shutdown to the large release frequency of the safety goals of 1E-7/yr.	
9.1	A safety analysis of the plant design shall include hazard analysis, deterministic safety analysis, and probabilistic safety assessment (PSA) techniques. The safety analysis shall demonstrate achievement of all levels of defence in depth, and confirm that the design is capable of meeting the applicable expectations, dose acceptance criteria and safety goals. Radioactive sources other than the reactor core, such as the spent fuel pool and fuel handling systems, shall be considered. Impacts for multiple units at a site if applicable, shall be included.	The requirements of this clause relevant to probabilistic safety analysis is covered in detail in the assessment of CNSC REGDOC-2.4.2.	RNA

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	The first step of the safety analysis shall be to identify PIEs using a systematic methodology, such as failure modes and effects analysis. Both direct and indirect events shall be considered in PIE identification. Requirements and guidance for identification of PIEs is given in section 7.4 of this document.		
9.2	The safety analysis shall be iterative with the design process, and result in two reports: a preliminary safety analysis report, and a final safety analysis report. The preliminary safety analysis shall assist in the establishment of the design-basis requirements for the items important to safety, and demonstrate whether the plant design meets applicable requirements. The final safety analysis shall: 1. reflect the as-built plant 2. account for postulated aging effects on SSCs important to safety 3. demonstrate that the design can withstand and effectively respond to identified PIEs 4. demonstrate the effectiveness of the safety systems and safety support systems 5. derive the OLCs for the plant, including: a. operational limits and set points important to safety b. allowable operating configurations, and constraints for operational procedures 6. establish requirements for emergency response and accident management 7. determine post-accident environmental conditions, including radiation fields and worker doses, to	The requirements of this clause relevant to probabilistic safety analysis is covered in detail in the assessment of CNSC REGDOC-2.4.2.	RNA

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	confirm that operators are able to carry out the actions credited in the analysis 8. demonstrate that the design incorporates sufficient safety margins 9. confirm that the dose and derived acceptance criteria are met for all AOOs and DBAs 10. demonstrate that all safety goals have been met Guidance The Class I Nuclear Facilities Regulations requires a preliminary safety analysis report demonstrating the adequacy of the NPP design to be submitted in support of an application for a licence to construct a Class I nuclear facility. A final safety analysis report demonstrating the adequacy of the design is required for an application for a licence to operate a Class I nuclear facility.		
9.5	The probabilistic safety assessment shall be conducted in accordance with the requirements specified in CNSC REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants. Additional information Additional information may be found in: • ASME/ANS, RA-Sa-2009, Standard for Level 1/Large Early Release Frequency PRA for Nuclear Power Plant Applications, La Grange, Illinois, 2009. • CNSC RD/GD-369, Licence Application Guide:	This clause is covered in detail in the assessment of CNSC REGDOC-2.4.2.	RNA

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	Licence to Construct a Nuclear Power Plant, Ottawa, Canada, 2011. • CNSC, REGDOC-2.4.2, Probabilistic Safety Assessment (PSA) for Nuclear Power Plants, Ottawa, Canada, 2014. • IAEA, SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants, Vienna, 2010. • IAEA, SSG-4, Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, Vienna, 2010. • IAEA, Safety Series No. 50-P-10, Human Reliability Analysis in Probabilistic Safety Assessment for Nuclear Power Plants, Vienna, 1995. • IAEA Safety Reports Series No. 25, Review of Probabilistic Safety Assessments by Regulatory Bodies, Vienna, 2002. • IAEA, Safety Series No. 50-P-7, Treatment of External Hazards in Probabilistic Safety Assessment for Nuclear Power Plants, Vienna, 1995. • IAEA, Safety Report Series No.10, Treatment of Internal Fires in Probabilistic Safety Assessment for Nuclear Power Plants, Vienna, 1998.		