FORM-14159 R000*

Periodic Safety Review - Final Document Review Traveler

Bruce Power

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Bruce Power Document Title:
Bruce B Periodic Safety Review Basis Document

Bruce Power Contract/Purchase Order:	Bruce Power Project #:
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A Report Submitted to Bruce Power

January 14, 2016

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Acronyms and Glossary

AI	Action Item
AR	Action Request
CA	Corrective Action
CAP	Corrective Action Plan
CARD	Corrective Action Requirements Definition
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
CSI	CANDU Safety Issue
FAI	Fukushima Action Item
GAF	Global Assessment Framework
GAR	Global Assessment Report
GIO	Global Improvement Opportunity
HMI	Human-Machine Interface
IAEA	International Atomic Energy Agency
IIP	Integrated Implementation Plan
ISR	Integrated Safety Review
LCH	Licence Conditions Handbook
MCR	Major Component Replacement
NPP	Nuclear Power Plant
OPEX	Operating Experience
PIO	Potential Improvement Opportunity
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment (synonymous with PRA)
PSR	Periodic Safety Review
RIDM	Risk-Informed Decision Making
SBR	Safety Basis Report
SCDF	Severe Core Damage Frequency
SCR	Station Condition Record
SFR	Safety Factor Report
SIP	Safety Improvement Plan

SSCs	Structures, Systems and Components

Corrective Action Plan (CAP): A work execution plan identifying the activities and tasks that need to be performed to meet the requirements in the corresponding CARD, together with the corresponding budget, schedule, and resource allocation.

Corrective Action Requirements Definition (CARD): An elaboration of a corrective action to facilitate development of a detailed Corrective Action Plan. The CARD comprises a statement of the objective of a specific corrective action, requirements for its successful execution, deliverables, as well as closure criteria which will enable verification of successful completion.

Global Improvement Opportunity (GIO): A single comprehensive expression that represents a collection of PIOs and/or Safety Factor Macro-Gaps that have been consolidated, aggregated and integrated to describe an overarching potential improvement opportunity.

Potential Improvement Opportunity (PIO): An improvement initiative that originates from the ISR or PSR process or that may already be planned or in progress having originated elsewhere. Other origins of PIOs include: the Safety Improvement Plan (SIP), CNSC Action Items, previous IIPs, and other capital projects.

Practicable: Both practical and justified by cost-benefit analysis.

Safety Factor Macro-Gap: A single comprehensive expression that represents a collection of Safety Factor Micro-Gaps that were grouped together in a Safety Factor Report because they are essentially the same or closely related or are perceived to have the same underlying cause.

Safety Factor Micro-Gap: A description of a clear case of a plant design feature or program not meeting the requirements of a specific clause or group of clauses of a modern standard or review element of SSG-25 or REGDOC-2.3.3 as expressed in a Review Task, or a gap in effective implementation of the design requirement or program.

1. Introduction

1.1. Purpose

This document is the Basis Document for the Bruce B Periodic Safety Review (PSR) and sets out the scope and methodology for the PSR to be conducted for continued safe and reliable operation of Bruce B for 10 years extending from 2016 to 2025. The PSR Basis Document will be submitted to the Canadian Nuclear Safety Commission (CNSC).

This basis document will govern the conduct of the PSR and facilitate its regulatory review to ensure that Bruce Power and the CNSC have the same expectations for the scope, methodology and outcomes of the PSR. The basis document includes the following elements:

1. the scope;
2. major milestones, including cut-off dates (beyond which changes to codes and standards and new information will not be considered);
3. the current licensing basis including exemptions and acceptable deviations;
4. the applicable national and international standards, codes and practices;
5. the operating strategy of the facility;
6. the methodology for the performance of the PSR, including the period for which the PSR is valid;
7. the methodology for the global assessment including the process for categorizing, prioritizing, resolving, and tracking findings;
8. description of the methodology for the identification, dispositioning and tracking of gaps;
9. the structure of the PSR documentation; and
10. PSR governance.

1.2. Background

As Bruce Power conducts long term operation activities including, refurbishment of reactor and replacement of major components for the six Bruce A and B units, there are requirements that it must meet. In accordance with Section 15.2 of the Power Reactor Operating Licence (PROL), Bruce Power is required to inform the Commission 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.

Due to the unusually long outage and de-fuelled state, there is also an opportunity to conduct work that could not be done in a regular maintenance outage. Bruce Power initiated its safety basis strategy with the submission of a Safety Basis Report in December 2013 [1]. This submission, which included an interim PSR, supported licence renewal and committed Bruce Power to the submission of full PSRs for each of Bruce A and Bruce B by the end of the next licensing period (e.g., 2020). Follow-up work to the Safety Basis Report was submitted in October 2014. Bruce Power continued this strategy with the implementation of the Bruce A Integrated Safety Review (ISR), which will be completed by the end of 2016.

The Bruce B PSR represents the second portion of the licence commitment made in the Safety Basis Report, as well as supporting the future Bruce B Major Component Replacement (MCR) outages and asset management. To achieve this, Bruce Power will conduct a review of safety for the future refurbishments of Bruce B that is consistent with the current Bruce A ISR. The Bruce B PSR will satisfy Licence Condition 15.2 – Continuous Operations of the current licence (PROL 18.00/2020) [2], which requires compliance with REGDOC-2.3.3 [3].[1] Furthermore, the Bruce B PSR will meet or exceed the international guidelines given in International Atomic Energy Agency (IAEA) Guide SSG-25 [5], Periodic Safety Review for Nuclear Power Plants. The project deliverables will build upon those from the Bruce A ISR, and will provide the foundation for the safety reviews for future PSRs at Bruce A and B. This safety review may also identify other opportunities for improving safety that can be done through the Bruce Power Corrective Action (CA) Program.

Results of the PSR will be submitted to the CNSC in the following documents:

1. PSR Basis Document (this document);
2. Reports on the review of each Safety Factor (SFR); and
3. Combined Global Assessment Report (GAR) / Integrated Implementation Plan (IIP) - Bruce Power will submit the combined GAR and IIP following review of the Safety Factor Reports by the CNSC.

1.3. Scope and Structure of Document

The PSR Basis Document builds upon the Bruce A ISR Basis Document [6]. The rest of this basis is structured as follows:

- **Section 2: PSR Scope** describes the objectives of the PSR and the overall scope of the PSR effort. It describes the current licensing basis and identifies the Safety Factors to be reviewed with reference to REGDOC-2.3.3 and refers the reader to Appendix A for a detailed exposition of the review tasks of SSG-25 and REGDOC-2.3.3. It describes the process for selecting the set of modern codes and standards to be used for these reviews and refers to Appendix B for the Bruce Power governance documents and Appendix C for the list of codes and standards used in the PSR [7].
- **Section 3: PSR Methodology** begins by describing the systematic approach to the PSR that will be followed before delving into the details of the three major phases of the PSR

[1] REGDOC-2.3.3 [3] has superceded RD-360, Life Extension of Nuclear Power Plants, [4]. In the Bruce A ISR, the basis document was based on the guidelines contained in CNSC RD-360.

process: Safety Factor reviews, Global Assessment, and Integrated Implementation Planning. The conduct of Safety Factor reviews is described in terms of the nature and scope of assessments against the provisions of modern codes and standards and how this will feed into the execution of the review tasks. The section on Global Assessment describes how the findings of the Safety Factor reviews will be consolidated and integrated to arrive at overall conclusions regarding the continued safe operation with the asset management and MCR strategy in both Bruce A and B. It also identifies 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. Finally, the section on Integrated Implementation Planning describes how improvement opportunities will be developed, ranked, prioritized and where necessary subjected to Risk Informed Decision Making (RIDM), including plans formulated to implement the selected improvements to arrive at an IIP.

- **Section 4: Recording the Output of the PSR** describes how a database will be used to support tracking of gaps and recording the results of the execution of the PSR, as well as the content of the main outputs of the PSR: SFRs, GAR and IIP.
- **Section 5: Managing the Execution of the PSR** describes Bruce Power’s organization for the PSR, the overall time-frame with major milestones as well as quality assurance provisions.

2. Scope of the PSR

2.1. PSR Objectives

The safety of Bruce B is regularly and thoroughly assessed through several processes that are part of the current licensing framework:

- Periodic updating of the Safety Report;
- Periodic updating of the Probabilistic Safety Assessment (PSA);
- Ongoing internal reporting and correction of deviations from the current licensing basis through Bruce Power's corrective action program;
- Regular and special regulatory inspections and follow-up actions to correct deviations identified by the CNSC;
- Ongoing interface with the CNSC to resolve issues identified in various forms of CNSC action items (AIs), as well as independent safety oversight; and
- A thorough review of safety and confirmation of the safety of the plant by the CNSC via the licence renewal process, which currently occurs on a five year frequency.

The overall objective of the Bruce B PSR is to help define a confined scope of work for the Unit 5 through 8 MCR outages and augmented by comprehensive asset management plans and other improvements that will provide practicable opportunities for enhancing safety. The look-ahead period for this PSR will cover a 10-year period, i.e., from 2016 to 2025. The ten-year period is in accordance with REGDOC-2.3.3 and SSG-25, and the specific dates are consistent with the freeze date of December 31, 2015 (see Section 2.5). It is recognized that the 10-year period cycle will not align with the licencing period and the nominal design life for Structures, Systems, and Components (SSCs). However, the continuous nature of Bruce Power processes is in place to revisit the PSR to ensure that they are current. Nuclear Safety is a primary consideration for Bruce Power and the management system will support the enhancement and improvement of safety culture and the achievement of high levels of safety, as well as business performance.

The specific objectives of the PSR are to:

1. Determine the extent to which the plant meets modern codes and standards and industry best practices;
2. Determine the extent to which the licensing basis will remain valid over the operating life of Bruce B (by definition, the licensing basis is always valid, so this objective from SSG-25 and REGDOC-2.3.3 is interpreted to mean the extent to which the plant currently meets new requirements that may become part of the licensing basis in the future);
3. Determine the adequacy of the SSCs and programs that are in place to ensure safe and reliable plant operation over the PSR period ; and
4. Determine the practicable improvements to be implemented to resolve any findings identified in the review and timelines for their implementation.

2.2. PSR Scope

Bruce Power's position is that the current and near-term safe operation of Bruce B is assured by the existing processes noted above. Therefore, the PSR is not a process to assess the current safety of Bruce B. Rather, the PSR is a process to assess the medium to long-term prospects for safe operation of Bruce B and to identify and develop a plan to implement safety improvements to further enhance safety in support of longer term operation. This position is consistent with SSG-25 (Clause 2.8), which recognizes that some countries have satisfactory processes to regularly review plant safety, and that in such cases a PSR may not be required. However, Bruce Power views ISRs and PSRs as complementary to the existing processes.

The scope of this PSR is to conduct a review of Bruce B that meets the elements given in CNSC REGDOC-2.3.3 [4]. As part of this, the PSR will include a comprehensive review of current licensing issues applicable to the safety factors such as ongoing CNSC issues, Action Items, Fukushima Action Items (FAI), CANDU Safety Issues (CSIs), and licence submissions

The PSR will make use of the following reviews conducted previously:

- Return to service of Bruce Units 3 and 4 (circa 2001) [8];
- Life extension of Bruce Units 1 and 2 (circa 2006) [9] [10] [11];
- ISR Basis Document [12], 14 SFRs [13] [14] [15], GAR/IIP [16], conducted circa 2008 for the proposed Unit 3 and 4 refurbishments;

- Safety Basis Report (SBR) and PSR for Bruce Units 1 to 8 (2013) [1]; and
- Bruce A ISR Basis Document [6] and Bruce A Safety Factor Review Reports [17] (2015).

These reviews covered many, if not all, of the same Safety Factors that are reviewed in the Bruce A ISR and will be reviewed in the Bruce B PSR. A full chronology of Bruce Power safety reviews performed prior to 2013 is provided in Appendix F of [18].

In addition, the work on this PSR shall build on the outputs of ISR conducted in 2014/2015 in support of MCR in Units 3 and 4, and during asset management activities to support ongoing operation of all four Bruce A units, including U0A:

- Bruce A ISR Basis Document [6],
- 15 SFRs [17], and
- Draft Bruce A Global Assessment Report and Integrated Implementation Plan.

The process for the conduct of PSR can therefore be summarized as the following:

- Performing assessments of the Bruce B plant and Bruce Power governance against a set of current and modern codes and standards;
- Using the results of the assessments, evaluate the Bruce B plant and Bruce Power governance against 15 Safety Factors prescribed in REGDOC-2.3.3;
- Performing a Global Assessment that consolidates and integrates the findings of the Safety Factor reviews into an overall assessment of safety together with a list of ranked potential improvement opportunities; and
- Developing a set of practicable improvement opportunities while taking into account previously developed station improvement plans to arrive at a single IIP for Bruce A and B.

This PSR Basis Document is an essential instrument that governs the conduct of the PSR. It ensures that Bruce Power and the CNSC have the same expectations for the PSR's scope, methodology and outcomes.

It is also noted that the PSR is complementary to, and does not replace, regulatory activities required and/or performed by the CNSC, including routine and non-routine regulatory reviews and inspections, mid-term reports, event reporting and investigations, or any other CNSC licensing and verification activities.

2.3. Safety Factors to be Reviewed

The safety reviews for Bruce B will be compiled in the 15 Safety Factor Reports shown in Table 1. REGDOC-2.3.3 [3] Section 3.3 identifies the 15 safety factors, invoking SSG-25 [2] for the first 14 Safety Factor Reports, and adding a fifteenth Safety Factor Report for radiation protection.

In SSG-25 [5], the IAEA identifies five Subject Areas that are further broken down into a list of fourteen Safety Factors that are considered to be the basic topics for consideration as part of a comprehensive assessment of overall plant safety.

In carrying out the PSR, each Safety Factor is comprised of review tasks as recommended in SSG-25 [5] and REGDOC-2.3.3 [3]. The review tasks for each Safety Factor are listed in Appendix A. The concept underlying the PSR is to use these review tasks to assess the current design, operation, and governing programs of the plant using the assessments of the modern codes and standards to develop statements of adequacy. Gaps, if any, will be identified and addressed in accordance with the process outlined in Section 2.5 of this document.

Further elaboration on the scope associated with each of the Safety Factors and corresponding review tasks comprising the PSR is provided in Appendix A. Note that SSG-25 [5] includes the qualifier that use of all identified review tasks does not guarantee full coverage of a Safety Factor. When developing the scope of Safety Factor reviews, the review tasks documented in Appendix A will be used as a starting point and modified as necessary to address all gaps determined in the codes and standards assessments of Bruce B design and operation.

In particular, the scope of the review tasks shall cover Systems Important to Safety as defined by Bruce Power in compliance with CNSC document RD/GD-98, “Reliability Programs for Nuclear Power Plants” [19] and systems defining the Safe Operating Envelope (SOE). It is to be noted that Bruce Power employs a number of SSC lists to serve specific objectives as related to different aspects of safety considered in, for example, design, safety analysis, equipment reliability, structural integrity. The most important and comprehensive of these is the Safety Related System List (SRSL). The list applies to all work related to the execution of design, commissioning, maintenance and operation of the systems.

Other lists with more specific safety-related purposes include:

- The System Classification List, which categorizes systems and components in accordance with their importance to nuclear safety with reference to the applicable section of the ASME code for design and construction purposes.
- The Environmental Qualification Safety Related Component List (EQSRCL), which is a list of all EQ safety-related equipment and components, including their harsh/mild indicators and evaluation for degradable materials.
- The Seismic Qualification of Safety-Related Systems Design Guide, which defines the safety-related systems requiring seismic qualification.
- The Fire Safe Shutdown Success Path, which is a list that identifies all of the SSCs credited for the safe shutdown of the plant in the event of a fire.
- The SOE system list, which identifies the systems for which the SOE Operational Safety Requirements apply. This list includes systems that are credited with a design basis accident mitigation function in the Safety Report or supplementary analysis, and includes systems where their initial conditions could impact on accident consequences. However, the systems on this list are not necessarily in the SRSL.

- The Assessment of Systems Important to Safety for the Safety & Licensing Portion of the Nuclear Asset Management Program, which presents the various system groupings at Bruce Power that rank the importance of SSCs based on safety and production.

Table 1: Scope of the PSR

Subject Area	Safety Factor
The Plant	1. Plant Design 2. Actual Condition of Systems, Structures, and Components 3. Equipment Qualification 4. Ageing
Safety Analysis	5. Deterministic Safety Analysis 6. Probabilistic Safety Analysis 7. Hazard Analysis
Performance and Feedback from Operating Experience	8. Safety Performance 9. Use of Experience from other Plants and Research Findings
Management	10. Organization and Administration (including Quality Management - CNSC recommended) 11. Procedures 12. The Human Factor 13. Emergency Planning
Environment	14. Radiological Impact on the Environment
Radiation Protection	15. Radiation Protection

The PSR shall also include a comprehensive review of the following licensing issues for the station in the codes and standards and review task assessments where they may indicate gaps:

- Safety Report Update issues;
- CANDU Generic Safety Issues, including CANDU Safety Issues (CSIs);
- Bruce Power Regulatory Commitments; and
- Tracking of licence concessions.

2.4. Statement of Current Licensing Basis

REGDOC-2.3.3 Section 3.1 requires that the PSR Basis Document include a description of the current plant licensing basis at the time of initiation of the PSR. The term “licensing basis” is defined in CNSC document INFO-0795 [20] as follows:

> *“The Licensing Basis for a regulated facility or activity is a set of requirements and documents comprising:*
>
> *(i) the regulatory requirements set out in the applicable laws and regulations*
> *(ii) the conditions and safety and control measures described in the facility's or activity's licence and the documents directly referenced in that licence*
> *(iii) the safety and control measures described in the licence application and the documents needed to support that licence application.”*

With respect to item (i), the applicable laws and regulations are primarily contained in the Nuclear Safety and Control Act (the Act) and the Regulations issued under the Act. These are available on the CNSC website at http://nuclearsafety.gc.ca/eng/acts-and-regulations/acts/index.cfm. The same website also lists other federal acts that may be applicable to nuclear facilities.

With respect to item (ii), the Licence Conditions Handbook (LCH) [21], Section 1.1, clarifies the licensing basis and states that first level references in the licence are part of the licensing basis. Second or lower level references are not part of the licensing basis, except for those items or sections thereof which are specifically referenced in the licence as distinct licence conditions. The documents directly referenced in the current operating licence [2] (the first level documents) are listed in Table C-1 (refer to the column headed “Revision in Licence or LCH”). The licence identifies applicable codes and standards by document number and title. The LCH provides additional information, including the revision date of the applicable code and standard. The LCH also identifies the date by which Bruce Power is expected to come into compliance with the codes and standards. In most cases, this "effective date" is the start of the licence period, but in cases where a new code or standard was introduced into the licence the effective date is partway through the current licence period.

The LCH also identifies additional codes and standards that are used in the compliance verification process, and describes how each fits into this process. In some cases, for example, compliance is expected. In other cases, a transition plan is expected. Table C-1 includes these additional codes and standards and their context as stated in the LCH.

With respect to item (iii), the Bruce Power governance documents that support the licence application are listed in Table B-1. These documents were extracted from the licence renewal application submitted in 2013 [22] in support of the current operating licence, from supplementary submissions in 2014 [23] [24], and from additional documents identified in the LCH [21] as being part of the licensing basis.

2.5. Statement of Modern Codes, Standards and Practices

This section describes the scope of the codes and standards applicable to the Bruce B PSR, and which modern codes and standards will be assessed.

One of the key elements of the PSR is the assessment of compliance with applicable codes and standards. The general purpose of assessment of the modern codes and standards against the plant design and operation is to identify any gaps between the current licensing basis and additional conditions relating to nuclear safety in modern codes and standards.

The starting point for developing the list of codes and standards to be assessed for the PSR is identifying those that are included in the licencing basis and relevant to the review tasks identified in Section 5 of SSG-25 [5] and Appendix A of REGDOC-2.3.3 [3]. These were adjusted as necessary for the unique circumstances of Bruce B and are included as Appendix A.

The next step was to identify a suite of codes and standards, the requirements of which generally cover the review tasks of each Safety Factor.

- The approach was first to identify the modern versions of the regulatory documents codes and standards in the licensing basis.
- The next step was to identify additional applicable Canadian standards (i.e., Canadian Standards Association (CSA)) that are not already included in the current licensing basis. Although not equivalent to "Codes and Standards," IAEA guides were also considered for applicability.

The general philosophy with respect to identifying the specific codes and standards to be used in the PSR was to apply the hierarchy described above, to avoid the need to perform redundant assessments. For example, the provisions of many IAEA documents are embodied in CNSC or CSA documents. In such cases, only the top document in this hierarchy was selected. On this basis, documents such as IAEA standards will only be specified for review if there is no current Canadian standard or accepted best practice. Versions of modern codes and standards issued for use by the freeze date[2] of December 31, 2015, are considered for review in this PSR.

Appendix C contains the list of the codes and standards to be assessed. Bruce Power already has a process to assure compliance with codes and standards, or to transition into compliance with codes and standards, in the PROL. Also, Bruce Power has already assessed a number of non-licence codes as part of earlier ISRs and PSRs. These assessments may be used as inputs to this PSR.

Accordingly, Table C-1 identifies the relevant codes and standards as either;

- Referenced in the PROL [2] so no assessments of these codes are required;
- Being transitioned into Bruce Power governance, therefore no assessment is required; or
- Not part of the PROL; therefore assessment is required.

[2] The terms "freeze date" and "cut-off date" are regarded as synonymous.

Gap assessments of the codes and standards in Table C-1 that have been previously been performed will be used in the PSR and their validity confirmed.

3. PSR Methodology

3.1. Introduction

The following serves as the basis for the Bruce B PSR methodology:

1. The codes and standards listed in Appendix C are the basis for this PSR. The starting point for this list is the final codes and standards identified in the Bruce A ISR;

   Guidance in Section 3.3 of REGDOC-2.3.3 states the following:

   "It is expected that the required effort to carry out a subsequent PSR of an NPP will often be considerably less than for the first; however, the subsequent PSR should consider explicitly if the earlier PSR conclusions remain valid (for example, in light of the time elapsed since it was performed)."

   These previous reviews are listed in Section 2.2, and covered many, if not all, of the same Safety Factors and associated regulations codes and standards that need to be reviewed. Further reviews of those that were previously reviewed will not be performed if their conclusions remain valid. Such cases have been identified in Appendix C.

   Modern revisions of some of the codes and standards have been identified in the licence renewal application and supplementary submissions for the current PROL [22] [23] [24]. Reference [24] identifies if there are transition plans for such codes and standards. Such transition plans will be used to address gaps in cases where a code or standard is included in the current licence, as was done in the Bruce A ISR.

2. Each Safety Factor review will address those elements listed in Appendix A. The Safety Factor reviews will include a review of the appropriate programs and their implementation for the review tasks. The review of program implementation will include a review of audits, self-assessments and other performance reviews that are available. In addition, relevant Station Condition Records (SCRs) will be collected and reviewed for existing deficiencies and planned corrective actions.

3. The Global Assessment and IIP will combine the results of the Bruce A ISR, prepared in 2015, with those in the current Bruce B PSR. The GAR/IIP will follow the methodology used for the draft Bruce A GAR / IIP prepared in 2015. In regards to the IIP, it will integrate the improvement opportunities identified in the current Bruce B PSR, the Bruce A ISR, with those submitted to the CNSC [25].

3.2. Systematic Approach to the PSR

The approach to the execution of the PSR is comprehensive and systematic, following a set of well-defined steps, each relying on successful completion of the previous one, while maintaining traceability to the results of all the previous steps.

The comprehensive aspect of the systematic approach refers to the fact that a wide net is cast when it comes to the assessment against modern codes and standards, while at the same time also recognizing that there is already a host of initiatives, planned or in progress, that have their origin in past assessments.

The step-wise sequential aspect of the systematic approach is described in detail in the sections that follow and is based on a bottom-up process for collecting, integrating, and consolidating of all issues to be addressed followed by a top-down ranking and planning process of corrective actions.

Starting with individual gaps found against codes and standards and review tasks by the various Safety Factor reviews, consolidation and integration progresses upward, firstly within each Safety Factor, then across all Safety Factors and finally, by inclusion of all other improvement initiatives initiated elsewhere, to a set of ranked Global Improvement Opportunities (GIOs). Next corrective actions are identified for each GIO, and ranking is applied to arrive at a final set of practicable corrective actions. Once the requirements needed for the successful corrective actions have been defined, implementation plans for each corrective action are developed.

The process is illustrated in Figure 1. The PSR is integrated with Asset Management (AM) and MCR initiatives, as illustrated in Figure 2.

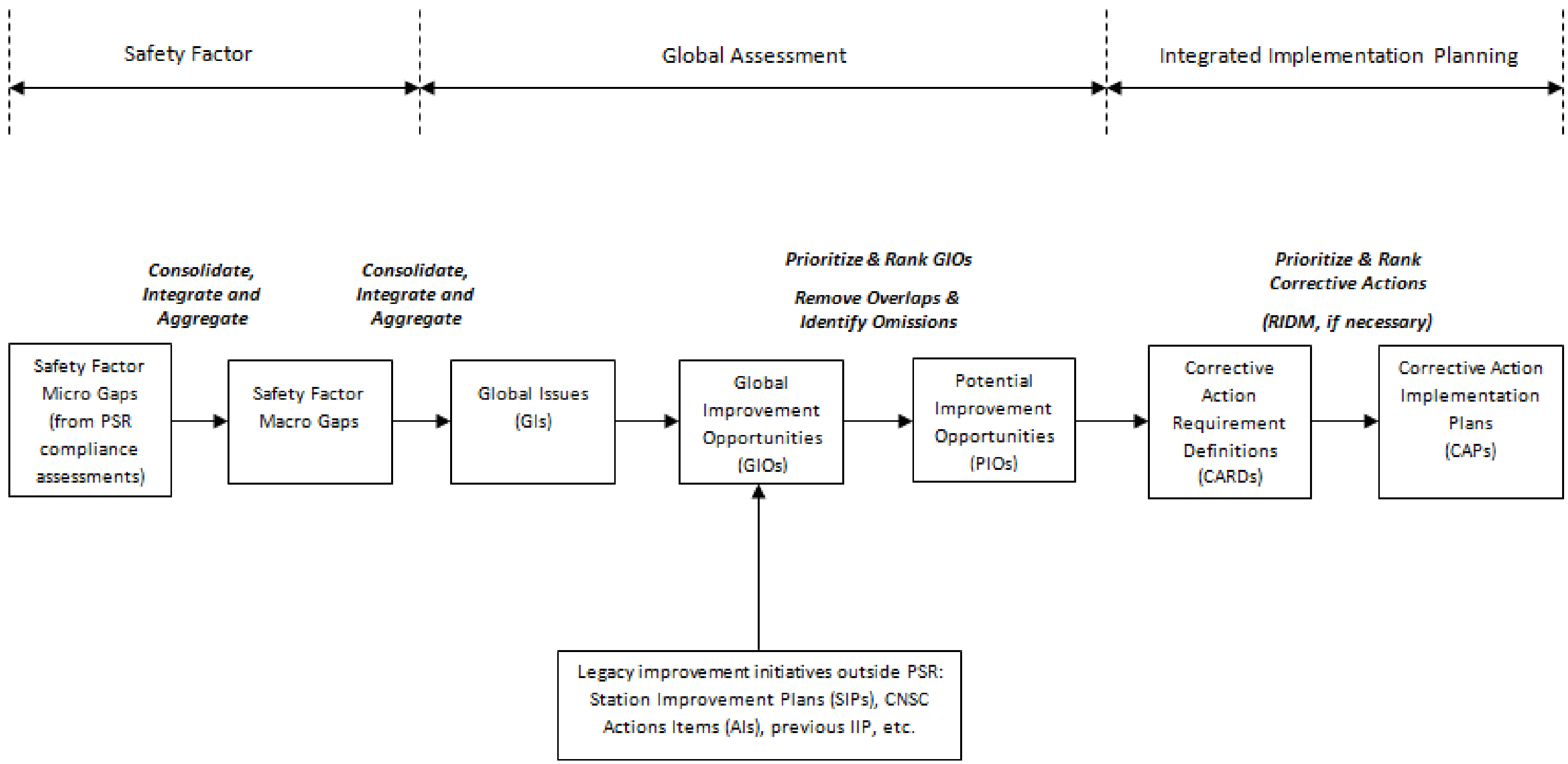


Figure 1: Periodic Safety Review Process

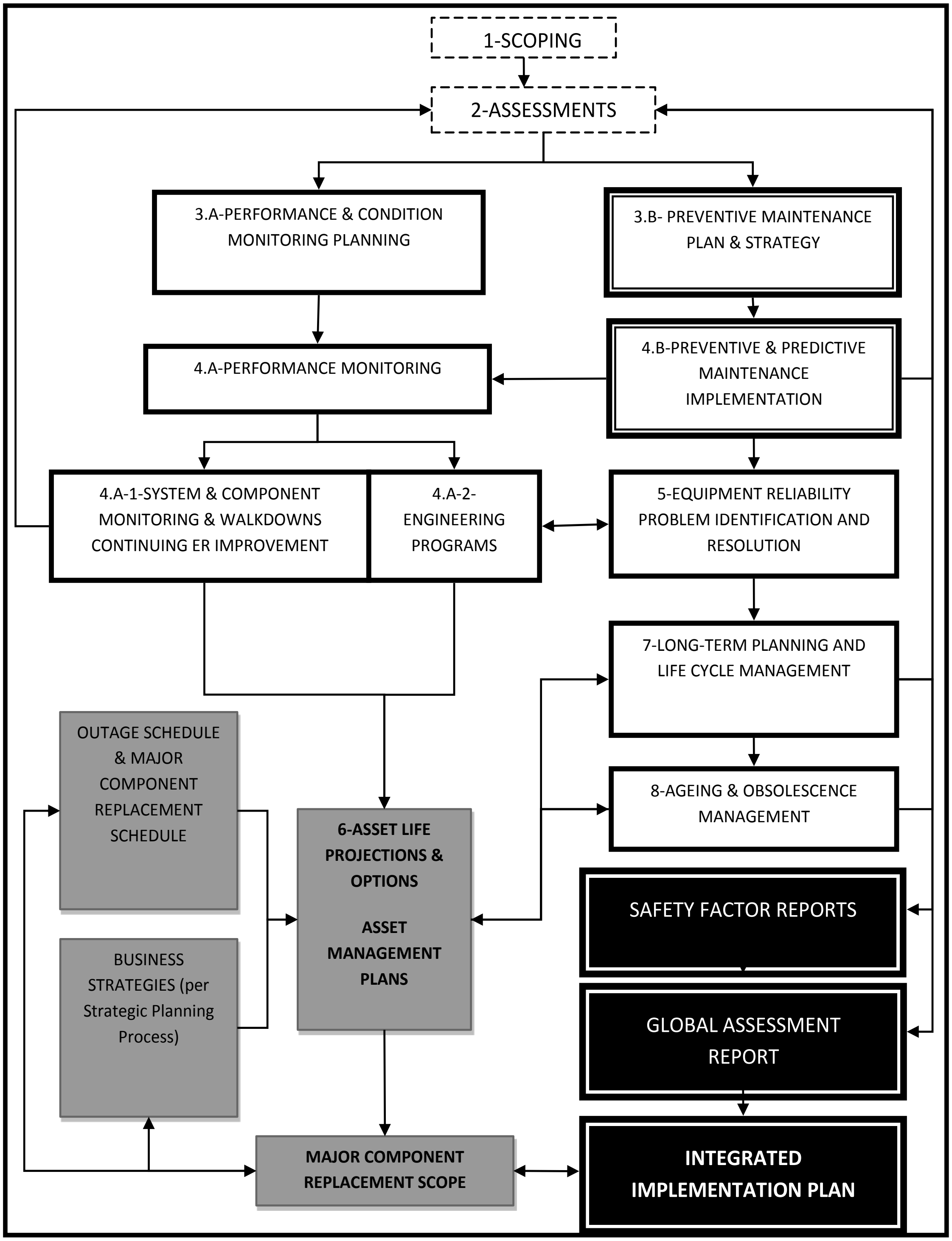
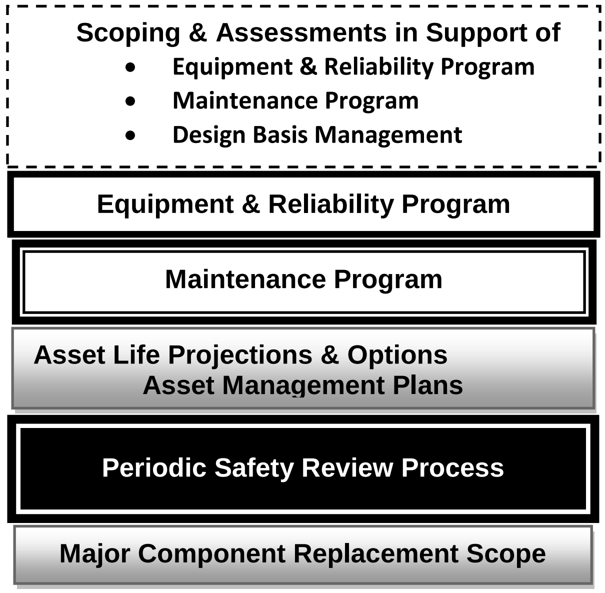


Figure 2: Relationship Between Equipment Reliability, Maintenance, Asset Life Management, Major Component Replacement and IIP

Figure 2 Legend:



3.3. Conducting Safety Factor Reviews

This generic review process of Safety Factors described in REGDOC-2.3.3 and SSG-25 has been tailored to proceed along the following steps for the Bruce B PSR:

1. Interpret and confirm review tasks;
2. Confirm the codes and standards to be considered for assessment;
3. Determine the type and scope of assessment to be performed for each code/standard;
4. Perform gap assessment against codes and standards;
5. Assess review tasks and align results of step 4 with the provisions of the review tasks;
6. Identify negative and positive findings; and
7. Document the assessment in SFRs.

Each of these steps is discussed in more detail below.

3.3.1. Interpretation and Confirmation of Review Tasks

As part of the preparation of this PSR Basis Document, the *Objective*, *Scope and Tasks*, and *Methodology* guidance provided in Section 5 of SSG-25 [5] has been adapted directly or adjusted as necessary for Bruce B, making use of experience from the Bruce A ISR [17]. The final set of review tasks is included in Appendix A.

As the first step in the Safety Factor Review, the review tasks are confirmed by the subject matter expert leading the review of the particular Safety Factor to ensure a common understanding of the intent and scope of each task. This confirmation may take the form of some elaboration of the review task to ensure that the reviews are conducted in a precise and specific manner.

The results of this step are documented in Section 2 (Methodology) of each SFR (see Section 4.2).

3.3.2. Confirm the Codes and Standards to be Considered for Assessment

The list of the codes and standards to be assessed is included in Appendix C. The subject matter expert for the Safety Factor shall validate the list against the defined review tasks to ensure that the assessment against each standard will yield sufficient information to complete the review task. In cases where the list of standards is inadequate, additional standards may be identified. 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 results of this step are documented in Section 3 (Applicable Codes and Standards) of each SFR (see Section 4.2).

3.3.3. Determine the Type and Scope of Assessment to be Performed

This step involves confirming that the assessment type for each of the codes and standards and guidance documents identified in Appendix C is appropriate based on the guidance provided below. This entails providing rationale for conducting an assessment as well as its type (Clause-by-Clause Plant, High-Level Plant, Clause-by-Clause Programmatic, High-Level Programmatic). In cases where it is only necessary to assess a subset of clauses, justification for the selection of clauses shall be provided.

The results of this step are also documented in Section 3 (Applicable Codes and Standards) of each SFR (see Section 4.2).

Gap assessments against codes and standards fall into two categories:

- **Programmatic Assessments** where the content of applicable programmatic documents such as policies, plans, and procedures are assessed against a single specific standard to determine the degree to which the program meets the provisions of a standard.
    - In Clause-by-Clause Programmatic Assessments the assessment is conducted against the individual clauses of the standard to demonstrate with supporting evidence whether the practices/requirements identified in the clause are met by Bruce Power. The review results shall be presented in a table with a compliance statement for each relevant clause. Any gaps against the clause shall be identified; and
    - In a High-Level Programmatic Assessment the scope of the relevant BP program is assessed to establish the degree of compliance with the modern versions of codes, standards, or guide it is intended to satisfy.

- **Plant Assessments** where the design configuration and condition of SSCs are assessed against a specific standard to determine the degree to which the equipment meets the provisions of a standard.
  - Clause-by-Clause Plant Assessments apply mostly when the standard specifies required design features or general requirements that apply at the system level. Clause-by-Clause plant assessments are applied to design or operation related codes and standards, identifying requirements or guidelines that could directly impact on the installed plant design and may impact on the design scope of asset management or refurbishment projects. A clause-by-clause review shall entail a review of each relevant clause in the code/standard to demonstrate with supporting evidence whether the practices/requirements identified in the clause are met by Bruce Power. Any applicable sub-tier referenced sections in the mandatory clauses to other codes, standards and Bruce Power documentation will also be addressed. The review results shall be presented in a table with a compliance statement for each relevant clause. Any gaps against the clause shall be identified; and
  - High-Level Plant Assessments apply to codes and standards for specific categories of components or programs, such as pressure-boundary components, and that provide detailed requirements for their design, construction and maintenance.

3.3.4. Perform Gap Assessments against Codes and Standards

This step provides for the actual assessment of the Bruce 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's design or programs meet the provisions of the specific clause of the standard or of some other criterion like a summary of related clauses. It is customary to use a table format in which the assessment text is entered against a verbatim reproduction of the clause (or summary) text in the same row. The assessment will be in the form of the assessor's reasoning as to why the clause is considered to be met or not met, while citing appropriate references that back up this contention. In the column next to the assessment a compliance indicator will be entered for each assessment based on the following compliance categories:

- 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 – the intent of the requirement is not met.

- Guidance – A potential programmatic, engineering, analytical or effectiveness gap found against guidance;
- Relevant but not Assessed (RNA) – the particular clause provides requirements that are less strenuous than clauses of another standard that has already been assessed. The definition also includes the guidance portion of clauses in which a gap has already been identified against the requirement;
- Not Relevant (NR) – The topic addressed in the specific clause is not relevant to the Safety Factor under consideration but may well be assessed under a different Safety Factor. In such cases appropriate reference to other Safety Factor Review(s) is provided.; and
- Not Applicable (NA) – The clause does not apply to the specific facility or the text is not a clause that provides requirements or guidance.

Notwithstanding the compliance categories above if, in the course of these assessments, it is found that the Bruce B plant or programs are potentially not in compliance with its current licence or if a gap is deemed to have high safety significance the issue will be brought to Bruce Power management's attention immediately. Such immediate risks found during the review will be addressed via Bruce Power's SCR process. The fact that such an event has occurred will be noted in the relevant Safety Factor Report and GAR.

It is customary to designate each individual deviation from the provisions of codes and standards as a Safety Factor micro-gap.

Finally the review task or tasks to which the assessment applies is also noted against the clause and assessment.

Newer versions of regulatory documents and standards often include guidance on how specific requirements may be met. The treatment of such guidance for the Bruce B PSR is based on consideration of the following:

- It is recognized that a reason for performing a PSR is to identify potential improvements that should be considered to enhance safety to support longer term operation. Such improvements may emanate both from mandatory requirements or guidance.
- Identification of a gap will not necessarily lead to an action to close the gap. Gaps get grouped, and then dispositioned through a process that considers aspects such as benefit and practicability.
- A proliferation of "gaps" flowing from assessments against guidance may be counterproductive in that it may distract attention from what is important and complicate the global assessment process unnecessarily.
- In cases where the guidance recommends a way to meet a mandatory requirement but the assessment shows that the requirement is met in a different way than suggested by the guidance, the Indirect Compliance category will be assigned;
- In cases where the guidance recommends additional practices over and above those given in requirements the assessment will clearly state that it is against guidance and not mandatory;

- In cases where guidance does not directly support the execution of Safety Factor review tasks, it not be assessed.
- A clear distinction will be made between gaps found against guidance and gaps found against requirements by assigning the gap sub-category: "Guidance" and prefacing the description of gaps related to guidance with the phrase: "Gap against guidance".
- In clause-by-clause assessments where guidance sub-clauses are intermingled with requirement sub-clauses in the same clause, both types of clause will be assessed and if any gaps are found the gap description will identify whether the gap was found against guidance or a requirement.
- A high-level assessment may be performed at the discretion of the assessor in cases where guidance is in the form a large or whole section of a document or an appendix.

In the majority of cases the Bruce B PSR will not be the first time that the Bruce B plant and programs have been assessed against codes and standards. The current assessment will therefore build on past assessments where possible and practical. The different scenarios for making use of past assessments are discussed below for the four review types.

3.3.4.1. Program Clause-by-Clause Assessments

Given any particular standard for which a programmatic clause-by-clause assessment has to be updated or confirmed there are four possible scenarios depending on whether the standard has been reissued since it was last assessed. The four scenarios are addressed in Table 2.

Table 2: Assessment Scenarios for Clause-by-Clause Programmatic Assessments

Scenario	Nature of Change in Standard	Assessment Approach for PSR
1	Compliance against the clauses of the standard was previously assessed and the standard has not been replaced by a new version before the freeze date of December 31, 2015.	Confirm that the previous assessment is still valid by updating references to programmatic documents.
2	Compliance against the clauses of the standard was previously assessed and the standard has been replaced by a new version in which some of the clauses have been changed before the freeze date of December 31, 2015 and the code or standard is not referenced in the current PROL or LCH.	Confirm that the previous assessment is still valid for unchanged clauses by updating references to programmatic documents. Assess compliance with updated clauses.

Scenario	Nature of Change in Standard	Assessment Approach for PSR
3	Compliance against the clauses of the standard was previously assessed and the standard has been replaced by a new version in which the original version has been completely rewritten and restructured before the freeze date of December 31, 2015 and the code or standard is not referenced in the current PROL or LCH.	Map the contents of the new version of the standard to the clauses of the previous version and apply past assessments that are still valid. Assess compliance with new requirements.
4	The standard is new having been issued before the freeze date of December 31, 2015 and a clause-by-clause assessment was not previously performed and the code or standard is not referenced in the current PROL or LCH.	Perform a complete clause-by-clause assessment.

The assessment scenarios for Clause-by-Clause Programmatic Assessments are also illustrated on the right hand side of Figure 3.

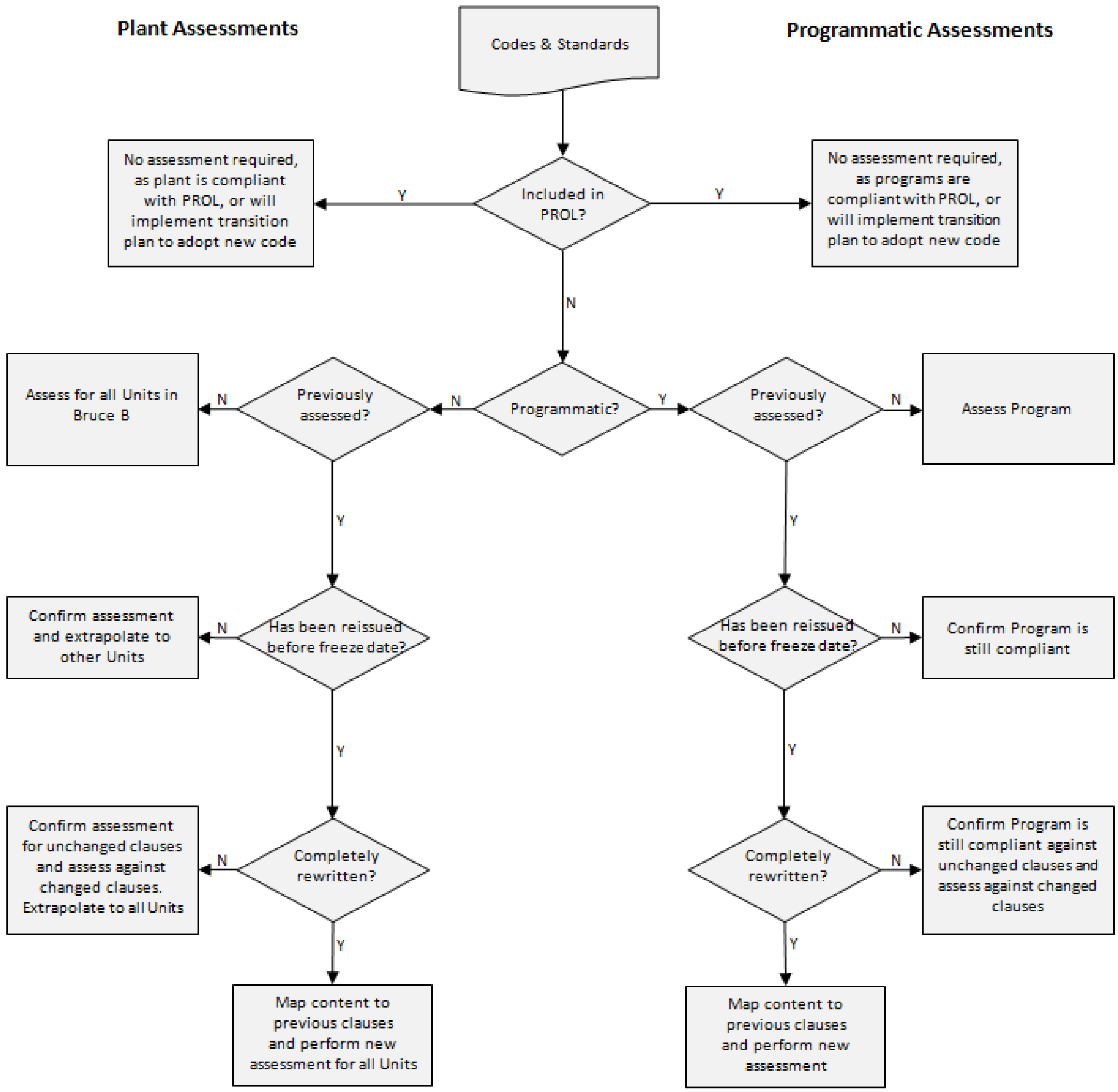


Figure 3: Assessment Scenarios for Clause-by-Clause Assessments

3.3.4.2. Plant Clause-by-Clause Assessments

Figure 3 also illustrates that, in principle, the same four assessment scenarios apply to Clause-by-Clause Plant Assessments. This is illustrated on the left hand side of the figure. The scenarios are further elucidated in Table 3.

Table 3: Assessment Scenarios for Clause-by-Clause Plant Assessments

Scenario	Nature of Change in Standard	Assessment Approach for PSR
1	Compliance against the clauses of the standard was previously assessed for some or all units and the standard has not been replaced by a new version before the freeze date of December 31, 2015.	Confirm that the previous assessment is still valid for the units for which it was performed. Identify configuration differences relative to other units at the time the earlier assessment was done and if necessary perform an assessment for the current configuration.
2	Compliance against the clauses of the standard was previously assessed for some or all units and the standard has been replaced by a new version in which some of the clauses have been changed before the freeze date of December 31, 2015 and the code or standard is not referenced in the current PROL or LCH.	Confirm that the previous assessment is still valid for the units for which it was performed. Identify configuration differences relative to other units at the time the earlier assessment was done and if necessary perform an assessment for current configuration. Assess compliance with updated clauses for all units.
3	Compliance against the clauses of the standard was previously assessed for some or all units and the standard has been replaced by a new version in which the original version has been completely rewritten and restructured before the freeze date of December 31, 2015 and the code or standard is not referenced in the current PROL or LCH.	Map the contents of the new version of the standard to the clauses of the previous version and apply past assessments that are still valid for the units for which it was performed. Identify configuration differences relative to other units at the time the earlier assessment was done and if necessary perform an assessment for the current configuration. Assess compliance with new requirements for all units.
4	The standard is new having been issued before the freeze date of December 31, 2015 and a clause-by-clause assessment has not been performed previously and the code or standard is not referenced in the current PROL or LCH.	Perform a complete clause-by-clause assessment for all units.

3.3.4.3. High-Level Program Assessments

The High-Level Programmatic Assessments involve two complementary aspects:

- A **Program-to-Standard Assessment** that compares the policies, programs, and procedures in place at Bruce Power against the codes and standards considered in the PSR. The results of this assessment are documented in Section 4 (Overview of Bruce Power Programs and Processes) of each SFR (see Section 4.2 herein).
- An **Implementation and Effectiveness Assessment** in which independent audits, self-assessments and regulatory evaluations and reviews pertaining to the particular program conducted in the last five years are identified and evaluated to determine the degree to which the program is implemented and meets its intent. The results of the assessment are documented in Section 7 (Program Assessment and Adequacy of Implementation) of each SFR (see Section 4.2).

Given that Bruce Power's programs apply corporate wide it follows that the results of previous assessments can be used for the Bruce B PSR provided that the results are updated to take account of any changes to these programs and their implementation during the intervening period. The following is therefore needed to update High-Level Programmatic Assessments:

- For Program-to-Standard Assessments:
    1. Determine whether any of the codes and standards cited in the original set of governing documents have been updated before the freeze date;
    2. Identify any structural changes in the hierarchy of documents that describe the program by identifying governing documents that have been added, removed, or updated; and
    3. Assess whether the current set of governing documents that make up the program addresses the latest set of codes and standards.
- For Implementation and Effectiveness Assessments: independent audits, self-assessments and regulatory evaluations that have been conducted in the last five years will be identified.

3.3.4.4. High-Level Plant Assessments

High-Level Plant Assessments apply to codes and standards that specify detailed requirements for the design, construction and maintenance of specific types of components like pressure vessels, piping, etc. These standards are often revised and updated to reflect the latest best practices. The purpose of High-Level Plant Assessments is to identify relevant updates to these codes and standards and to evaluate them for applicability so their provisions if practicable can be included in the design, construction, and maintenance processes for new or replacement components. Unless otherwise specified, existing equipment meets the provisions of codes and standards that were applicable at the time of construction.

To identify relevant changes to applicable codes and standards that are not in the PROLs, publications such as the Reedy Engineering Code Reconciliation Report [26] will be used.

3.3.5. Assess Alignment with the Provisions of the Review Tasks

The results of the gap assessment against codes and standards have to be interpreted in the context of the review tasks of the Safety Factor. To this end each assessment, whether clause-by-clause or high-level, will be assigned to one or more of the review tasks where applicable. Assessment against the provision of the review task involves using the related code assessment results to formulate a summary assessment of the degree to which the plant or program(s) meets the objective and provisions of the particular review task. This assessment may involve consolidation and interpretation of the various gap assessments to arrive at a single compliance indicator for the objective of the review task as a whole.

The results of this step will be documented in Section 5 (Results of the Review) of each SFR (see Section 4.2).

3.3.6. 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 will only be identified if there is supporting 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 will be designated as a Safety Factor micro-gap for tracking purposes. Identical or similar micro-gaps will be consolidated into comprehensive statements that describe the deviation topically known as Safety Factor macro-gaps. Macro-gaps will be listed in Section 8 (Summary and Conclusions) of each SFR (see Section 4.2). Macro-gaps will be prioritized and categorized at the Global Assessment and IIP stage.

3.3.7. Document the Assessment in Safety Factor Reports

The results of each Safety Factor assessment will be documented in a separate Safety Factor Report, one for each of the Safety Factors. The content of the SFR is described in Section 4.2.

3.4. Performing the Global Assessment

The Global Assessment (GA) methodology describes how the assessments of the following elements are conducted:

- The Global Assessment Framework (GAF) is developed to devise a systematic methodology and establish a common basis for assessing the relative importance of addressing global issues in terms of aspects, such as their safety significance and impact of their resolution. The same framework is used to assess the importance of practicable improvements and associated corrective actions.

- Integration of the results of the SFRs, in particular, the findings (both gaps and strengths) of nuclear power plant (NPP) design and operation in terms of overlaps, omissions, and interface issues;
- Assessment of interdependencies between integrated gaps, their classification, consolidation into global issues and the safety significance of their aggregate effects;
- Definition of potential improvement opportunities and recommended corrective actions for safety improvements to address individual and consolidated gaps are developed;
- Assessment of the extent to which the safety requirements of defence-in-depth are fulfilled;
- Estimation of global risk associated with facility operation with any unresolved gaps;
- Integration of improvement opportunities resulting from GA and on-going initiatives through an IIP; and
- A final report summarizing the results of the SFRs, GA and the associated IIP.

3.4.1. Consolidation of Safety Factor Findings

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

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

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

- Individual negative findings are defined as “Gaps[3]”.
- Collection of negative findings grouped and numbered in Section 8 “Summary and Conclusions” of each SFR are defined as “Macro-Gaps”.

Gaps across all SFRs will be uploaded in the ISR/PSR Database. Each gap will be provided a unique database identification. The following information is to be included for each gap:

- SFR Number
- Macro-gap Number and Title (as applicable)
- Reference to review task summary section and/or Appendix where the gap is identified
- Reference regulation, code or standard

[3] In general, the term ‘Gaps’ is used when Micro-Gaps are implied.

  - Applicable section or clause
  - Text of the requirement relevant to the gap
- Description of the gap
- Type of the gap (requirement, guidance, etc.)

3.4.1.1. SFR Gap Consolidation

The purpose of this step is to consolidate individual gaps or macro-gaps across all Safety Factor Report findings by establishing common features, thereby eliminating duplication. A "requirement based" comparison in terms of applicable regulations codes and standards and review tasks forms the basis for gap consolidation. Bruce Power governance is "program/process based", i.e., topical and as such in many cases the same requirement and associated gap may appear in more than one governing document. Hence, it may not always be feasible to consolidate gaps based on a requirement, but sometimes it may be more feasible to consolidate based on a program/process related topic.

Therefore, the approach used is to consider two aspects, first from a requirement perspective, then secondly from a program/process perspective. Hence, consolidation is implemented in two steps.

Step 1:

1. Starting with the first requirement where a gap is identified, scrutinizing the remainder of the gaps for coverage of the same requirement by using the Table of Gaps from the SFRs;
   a. If similar gap(s) or duplication is found:
      i. Linking the affected gaps; and
      ii. Identifying which gaps are duplicates or related across all affected Safety Factors.
   b. If no similar gap(s) or duplication is found, identifying remaining gaps for potential consolidation in Step 2.

Step 2:

2. Starting with the first remaining gap, scrutinizing the remainder of the gaps for coverage of the same topic or process;
   a. If similar gap(s) or duplication is found:
      i. Linking the affected gaps; and
      ii. Identifying which gaps are duplicates or related across all affected Safety Factors.
   b. Identifying any gaps that are unique.

This review will also help in forming a list of gaps that are topically similar which can be used in development of GIOs and Potential Improvement Opportunities (PIOs) in Sections 3.4.5 and 3.4.6.

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

The purpose of this step is to assess consolidated gaps from the SFRs to determine if they should be considered as PIOs as part of the IIP development and group them into Global Issues based on their topical similarities. Input for this step is the results of Section 3.4.1.

Gaps are assessed for their safety significance and priority as part of the GA and for consideration in the IIP. Section 3.6 of REGDOC-2.3.3 states the following:

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

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

> *5.10. Negative findings should be divided into:*
>
> - *Deviations for which no reasonable and practicable improvements can be identified;*
> - *Deviations for which identified improvements are not considered necessary;*
> - *Deviations for which safety improvements are considered necessary.*
>
> *5.12. In the case of negative findings for which no reasonable and practicable improvements can be identified, the reason(s) should be documented and the issue revisited after an appropriate period of time to determine whether a practicable solution is available. For negative findings for which safety improvement are not considered necessary, the reason(s) should be documented and the action considered completed. Negative findings for which safety improvements are necessary, including updating/or extending of plant documentation or operating procedures, should be categorized and prioritized according to their safety significance. The categorization and prioritization of safety improvements may be performed on the basis of deterministic analyses, probabilistic safety assessment, engineering judgment, etc. Safety improvements from the safety factor reviews, together with safety improvements resulting from the global assessment, should be included in the operating organization's integrated implementation plan.*

3.4.2.1. Assessment and Classification Scheme

Each gap is assessed and then classified under one of the three groups based on SSG-25 [5] articles 5.10 and 5.12. One additional category includes those gaps where there may already be an initiative to resolve it. For example, there may be an initiative in progress based on the previous IIP. These categories are described in the following sub-sections. The process is represented in Figure 4: Assessment and Classification Process.

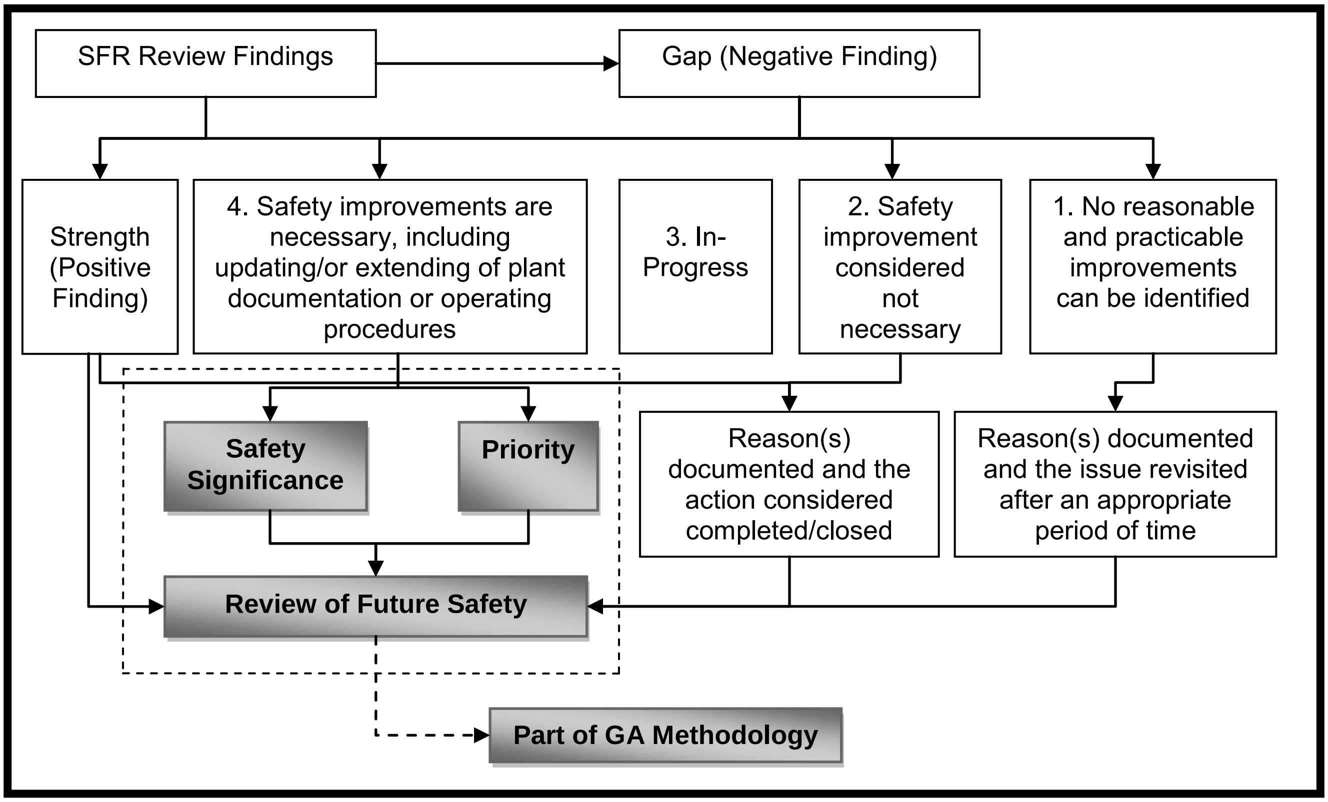


Figure 4: Assessment and Classification Process

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

Gaps in this category could generally result from comparison against modern codes and standards and some international practices that have not been incorporated in the Licencing Basis of the plant as prescribed in the PROL. Some examples are:

- A generic requirement which results in fundamental design changes to SSCs of the plant as a whole which cannot be accommodated within the current configuration of SSCs and plant layout, for example:
    - Higher safety margins than those currently in place due to an updated requirement.
    - New requirements which were not considered in the original design of the plant, e.g., physical changes driven by evolving design philosophy for new NPPs as additional requirements or newer interpretation of principles, such as redundancy, diversity, separation in terms of defence-in-depth (DID) or improvement of safety goals.

- Changes to the classification of SSCs which lead to different or new design requirements as compared to the current design basis of SSCs.

- A practice that is not adopted by either the CNSC or the Industry in Canada.
- A requirement that is not adopted by either the CNSC or the Industry in Canada that fundamentally impacts the organization of the plant, its governance and processes which is not sustainable in terms of business objectives.

For gaps in this category reason(s) for the classification are documented and the issue is revisited after an appropriate period of time (for example at the next PSR).

Integrated impact of not implementing these gaps is addressed as part of the Global Assessment.

3.4.2.1.2. Category 2: Safety Improvement Considered Unnecessary

Those gaps that do not have an impact on improving safety are considered as unnecessary to implement. Gaps in this category would generally result from:

- Comparison against modern codes and standards and some international practices where there are alternative ways of addressing them within the current licensing framework and industry best practices. In some cases contribution of additional provisions afforded by addressing such a gap would be rendered unnecessary because the current DID provisions and level of safety may be sufficiently robust and its contribution to overall safety goals may be insignificant. For gaps in this category, reason(s) for the classification are documented and the issue is categorized as “Closed” in the database.
- Individual gaps resulting from less than adequate implementation of the current governance and associated procedures. These are mostly identified during the review of audits, FASAs, peer reviews and in most cases specific corrective actions have already been identified for addressing them are in progress. In this context, they do not present a generic process improvement opportunity and can be dealt with through the current processes in place as appropriate. For gaps in this category, reason(s) for the classification are documented and a list is provided to Bruce Power for their consideration to decide if additional oversight is required outside the ISR process. Such gaps are categorized as “Closed” in the database.

3.4.2.1.3. Category 3: Safety Improvement In-Progress

Gaps that are the same as those that have already been identified in the previous ISRs or by other means are included in this category if there are initiatives or commitments in place to resolve them. Each gap is checked against the current IIP [25] initiatives or commitments and status of the corrective action in place is investigated and documented. Each review will result in one of the three sub-categories for such gaps:

- If the associated corrective action(s) is completed, appropriate references pertaining to the completion is provided and the issue is considered as “Closed”.
- If the associated corrective action(s) is in progress and being reported to the CNSC as part of the current IIP, appropriate references pertaining to the status is provided and the issue is considered as “Closed”.
- If the associated corrective action(s) is in progress but not being reported the CNSC as part of the current IIP [25], appropriate references pertaining to the status is provided and the issue is considered as “In-Progress”. An example would be a CNSC Action Item that is already in place but not reported as part of the current IIP. Such gaps are identified based on the list of non-SFR improvement initiatives described in Section 3.4.4 of the GA and included in the IIP directly.

3.4.2.1.4. Category 4: Safety Improvement Considered Necessary

This category includes the remaining gaps from Categories 1, 2 and 3 described above. These gaps are those where safety improvements are considered necessary. Generally these include:

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

3.4.3. Developing an Assessment Framework

The objective of developing an assessment framework, as described in Appendix D, is to devise a systematic methodology and establishing a common basis for assessing the relative importance of addressing global issues in terms of aspects such as their safety significance. The same framework will be used to assess the importance of practicable GIOs.

3.4.3.1. Development and Ranking of Global Issues

The purpose of this step is to develop and rank GIs making use of the insights gained in Section 3.4.1. Individual gaps identified in Section 3.4.2.1.4 are reviewed against each other for common features, thereby grouping them as a larger set as Global Issues. The approach used is to review two aspects of each gap; first from common or similar requirement(s) perspective and if no gap(s) are identified then secondly from a common process perspective. The approach is implemented in two steps.

Step 1: Requirement Based Grouping

1. Starting with the first requirement where an SFR gap is identified, scrutinizing the remainder of the gaps for coverage of the same requirement by using the results of Section 3.4.1;
2. If similar gap(s) associated with a requirement is found:
    a. Linking, in the database, the affected gaps to a single GI including all duplicates for completeness; and
    b. Providing a clear GI title and description that is as specific as possible to ensure it covers the associated gaps across all affected Safety Factors.
3. If no similar gap(s) is found, identifying remaining gaps to be considered for step 2 of the process.

Step 2: Process Based Grouping

1. Starting with the first remaining SFR gap, scrutinizing the remainder of the gaps for coverage of the same issue by using the results of Section 3.4.1;
    a. If commonality is found:
        i. Linking, in the database, the affected gaps to a single GI including (if any) duplicates for completeness; and
        ii. Providing a clear GI title and description that is as specific as possible to ensure it covers the associated gaps across all affected Safety Factors.
    b. If no commonality is found:
        i. Creating a GI for the individual gap; and
        ii. Providing a clear GI title and description that is as specific as possible

At the end of this step, all gaps are mapped to a GI. Each GI will be ranked at Tier 2 of the Value Tree as described in Appendix D.

Development of GIs is performed using the ISR/PSR Database. A specific verification shall be performed to ensure that all gaps are linked to a GI.

3.4.4. Establish Non-SFR Improvement Initiatives Outside of ISR/PSR: CNSC Action Items and Commitments, MCR Outage Scope

The purpose of this step is to collect and integrate all non-SFR improvement initiatives that have been identified through other assessments or initiatives outside the ISR/PSR that are related to list of consolidated gaps developed in Section 3.4.2, as well as other improvement initiatives to be considered in the IIP based on input from Bruce Power.

The list of non-SFR improvement initiatives provided by Bruce Power will include, but not be limited to, those originating from the following past assessments and ongoing activities appropriately cross-referenced to their original sources:

- MCR List of Initiatives;

- Capital Projects;
- CNSC Action Items;
- FAIs;
- CSIs; and
- Licence submissions

As part of step 3.4.2.1.3, the list of non-SFR improvement initiatives are reviewed against each gap to establish if there are any initiatives that, when completed, will support resolution of the issue associated with the gap. Such initiatives are linked to the associated gap(s) in the ISR/PSR database.

Those initiatives that are not related to any of the gaps as part of step 3.4.2.1.3, but to be considered in the IIP based on input from Bruce Power will be listed as non-SFR improvement initiatives to be considered in the development of the GAR and IIP.

The result of this step is a list of consolidated gaps with their associated improvement initiatives (if any) as part of sub-group 3 in step 3.4.2.1.3 and additional initiatives to be considered in the IIP based on input from Bruce Power.

The approach used here allows Bruce Power to augment the list of non-SFR improvement initiatives to be considered for the GA and IIP on a continuous basis, reflecting Bruce Power's long-term plans for safe and reliable operation of Bruce B beyond the current PSR or PROL.

3.4.5. Development of GIOs (Global Improvement Opportunities)

Collections of gaps and other initiatives from Section 3.4.4 will be integrated with GIs developed in Section 3.4.2 under entities known as GIOs. These have been defined as GIOs as some of the gaps in GIs will have been associated with planned initiatives based on Section 3.4.4, which mean that issues have already been recognized as improvement opportunities independent of this PSR. The approach used is the same as that used for development of GIs in Section 3.4.3.1. This step may involve expansion of the GI list due to integration of relevant initiatives identified in Section 3.4.4. Each gap and initiative will be reviewed against others to establish commonalities. If an initiative cannot be integrated with an existing GI, a new GIO will be created.

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

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

3.4.6. Development of PIOs for Consideration in the GAR and IIP

The purpose of this step is to consolidate, within each GIO, gaps from the SFRs and initiatives from other sources that are related to each other or that overlap in intent and/or scope and define them as PIOs. PIOs are used to help define CAs for developing and optimizing the IIP. PIOs contain consolidated gaps and relevant initiatives so that duplication of CAs are eliminated. Each PIO will be given a title and associated gaps and initiatives will be linked in the ISR/PSR database. Development of PIOs is performed using the ISR/PSR Database. A specific verification shall be performed to ensure that all gaps and initiatives are linked to a PIO within each GIO. It should be noted that steps 3.4.3.1, 3.4.4, 3.4.5 and 3.4.6 are iterative, such that individual gaps or initiatives can be moved between GIOs and PIOs to optimize development of corrective actions per Section 3.4.8.

One important aspect of PIO definition is that all gaps and initiatives consolidated under each PIO must be chosen such that they can be mapped under a single sub-objective (Tier 3) of the value tree described in Appendix D to ensure that relative ranking and prioritization of their CAs, to be developed in Section 3.4.8, can be performed in a consistent manner.

3.4.7. Prioritization and Ranking of GIOs

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

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

3.4.8. Develop Corrective Actions

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

The development of the CAs adheres to the following principles:

- An integrated approach to remove scope overlaps and optimizes available time and resources – corrective actions identified either through the SFRs or the other sources are integrated and consolidated;
- Deterministic and probabilistic safety assessment insights (e.g., where applicable, contribution to Severe Core Damage Frequency (SCDF) or safety goals or reduction in public dose, etc.) are utilized to the extent practicable in establishing risk importance, prioritization, and ranking of improvement opportunities that could be subject to a RIDM process (if necessary);
- Contribution of CAs to DID and the fundamental safety functions are taken into consideration;
- CAs to be taken for the implementation of each improvement opportunity are evaluated in terms of their contribution to actual benefit to safety taking into consideration how soon it will be effective once implemented;
- Alternative means of achieving the safety benefit are considered if adequate interim measures can be implemented that are commensurate with the safety significance of the PIO. ‘Do nothing’ will be considered as one of the options in all cases;

Initially, each PIO is converted to a corresponding CA to the extent possible. It is recognized that grouping PIOs together into a single CA may address more than one PIO, or it may be necessary to devise more than one CA for the same PIO. All PIOs of a GIO will have been mapped and defined in terms of CAs at the end of this step.

As for PIOs in Section 3.4.6, one important aspect of CA definition is that all gaps and initiatives consolidated under each CA must be chosen such that they can be mapped under a single Tier 3 sub-objective of the value tree described in Appendix D so that their relative ranking and prioritization can be performed in a consistent manner.

3.4.9. Prioritization and Ranking of Corrective Actions

The prioritization and ranking of CAs uses the GAF and follows the same process as that of GIOs, the only difference being that CAs will be assessed against Tier 3 of the value tree. Using the Global Assessment Framework described in Appendix D, as implemented in the database, the ranking and prioritization step entails the following:

- Associate each CA with the Tier 2 objective in the value tree that corresponds to the branch associated with the highest ranked GIO it is intended to address;
- Assign the CA to the appropriate Tier 3 sub-objective that the CA will support under the same second tier branch. In so doing, the CA assumes the same priority as the sub-objective as expressed in the weight of the sub-objective;
- Taking into consideration the nature of the CA use the GAF to evaluate the impact and time-to-take-effect of resolving the GIO. In so doing, a two parameter utility score is assigned to the CA;

- Calculate final score for the CA by multiplying the assigned weight from the 2nd bullet and utility score from the 3rd bullet; and
- Arrange the CAs based on final scores obtained in the 4th bullet from highest to lowest to arrive at a ranked list.

3.4.10. Perform Risk Informed Decision Making (as needed)

The need to perform a RIDM assessment will be established in consultation with Bruce Power and based on the scope, schedule and cost considerations associated with each CA. For example, a RIDM assessment would be required in cases where:

- The schedule for completion extends beyond planned outages or the MCR outage; or
- The associated costs are so extensive that implementation of similar or higher ranked CAs may be delayed; or
- Other considerations, such as Bruce Power's asset management plan expectations.

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

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

3.4.11. Global Assessment

The set of global strengths and global issues resulting from the consolidation of Safety Factor findings, the non-SFR improvement initiatives, and the identified practicable CAs forms the basis for the global assessment. The global assessment involves the formulation of arguments that seek to justify a position that it will be safe to continue operating the Bruce B station through MCR and asset management for life extension. It is to be noted that there will be a single Bruce A and B Global Assessment Report produced; i.e., the draft Global Assessment Report produced in the Bruce A phase of the project will be augmented to include the relevant Bruce B information. This formulation will therefore address the following:

- A global assessment based on the aggregate effect of the findings resulting from all SFRs, taking the proposed CAs and non-SFR improvement initiatives into account, together with their relative importance as expressed by their ranking numbers; and
- An assessment of the overall acceptability of operation of the NPP over the applicable period of the PSR and over the longer term outlook period.

3.4.12. Preparation of the Global Assessment Report

The results of the Global Assessment will be documented in a GAR. The content of the GAR is described in Section 4.3.1. The GAR will be combined with the IIP in a single report.

3.5. Integrated Implementation Plan

3.5.1. Purpose

The purpose of the IIP is:

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

3.5.2. Scope

The IIP will include the following activities and processes:

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

The CAs identified during the Global Assessment may be new, or may involve previously identified or ongoing activities, such as those included in the IIP for 2014 [25]. The objective of integrated implementation planning is to arrive at a single comprehensive set of cost-effective improvement initiatives that eliminates duplication and provides for maximum synergy.

3.5.3. IIP Methodology

The development of the IIP entails the following steps:

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

CAs in the development of the IIP adhere to the following principles:

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

It is noted that ranking and prioritization described in Section 3.4.9 is still valid for the High Level CAs and hence need not be repeated as long as the corrective actions associated with each CA is not integrated with another CA or changed. In such cases, ranking and prioritization of the integrated CAs will be performed as described in Section 3.4.9.

3.5.3.1. Develop a High Level Plan for Corrective Actions

The input for this step is a set of prioritized and ranked CAs from Section 3.4.9. Preparation of the plan includes:

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

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

3.5.3.1.1. High Level Scope

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

The high level scope will identify:

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

3.5.3.1.2. High Level Schedule

A high level schedule is based on input from Bruce Power. This high level schedule includes:

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

In order to minimize potential duplication and the effort associated with preparing high level implementation plans, Action Tracking ARs, outage or project plans and similar documentation that are already in place must be used. Such documents can be used as the high level implementation plan when deemed appropriate.

3.5.4. Optimize and Document the IIP

3.5.4.1. Optimize the IIP

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

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

3.5.4.2. Document the IIP

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

- An IIP in the form of proposed list of safety improvements, including their safety significance, prioritization and timing for implementation.
- The IIP is listed according to the CNSC's safety and control areas so as to facilitate the CNSC's review.
- To ensure success the IIP specifies:
    - Organizational arrangements in place to execute the IIP;
    - Governance applicable to the delivery of the IIP;
    - Where necessary, scope, schedules and dependencies, for the earlier tasks that have an impact on critical path;

- A high level definition of resources and a resourcing plan if constraints are specified with respect to availability of resources;
- The mechanism for overall integration, review and oversight; and
- Reference to a procedure that will govern change control of the IIP, or change control principles that will subsequently be incorporated into an IIP change control procedure.

3.5.5. Perform Global Assessment

The set of global strengths and global issues resulting from the consolidation of safety factor findings, the non-SFR improvement initiatives, and the identified practicable Corrective Actions in the IIP forms the basis for the Global Assessment. The GA involves the formulation of arguments that seek to justify a position that it will be safe to continue operating Bruce B for the PSR period and for life extension. This formulation will therefore address the following:

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

3.5.5.1. Assessment of Defence-in-Depth

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

SRS-46 [28] describes a method for assessing defence-in-depth capabilities of an existing plant, including both its design features and the operational measures taken to ensure safety. A broad spectrum of provisions, which encompass the safety features, equipment, procedures, staff availability, staff training and safety culture aspects, is considered. However, the PSR process also encompasses a systematic evaluation of the same aspects of defence-in-depth in an NPP using a different topical approach. In this context, SRS-46 states the following:

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

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

Table 2 of SRS-46 shows assignment of safety principles in INSAG-12 Basic Safety Principles for Nuclear Power Plants [29] to each level of DID. This relationship has been used as the basis for establishing an approach to the assessment of DID, as follows.

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

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

The following process is used:

1. Establish the applicable safety principles for the DID review.
2. Define DID levels impacted for each applicable safety principle in SRS-46 (taken from INSAG-12 [29]).
3. Map each safety principle to the relevant Safety Factor reviews that have been conducted.
4. Assess the DID aspects of each safety principle in the Bruce B design and operation at a high level. Compile and summarize associated evaluations from relevant Safety Factor Reports, as well as the Bruce B Safety Report Part 1: Plant and Site Description and Part 2: Plant Components and Systems as principal sources of information. Review results of SFRs for strengths, GA and IIP to determine those strengths and improvement initiatives that will demonstrate and further enhance alignment of Bruce B design and operation with the relevant safety principle.
    a. Summarize those features of the current plant design and operation that address the safety principle at a high level.
    b. Provide a list of SFR strengths, GIOs and associated CARDs included in the IIP that further improve alignment with the safety principle.
5. Provide an overall summary integrating conclusions from each step above for each level of DID.

3.5.5.2. Assessment of Overall Risk

Assessment of overall risk will be addressed in terms of:

1. **Significant improvements implemented in Bruce B**

   In this sub-section a summary of major projects undertaken to improve the physical plant to meet PROL conditions and to confirm safety margins as they relate to the deterministic and probabilistic safety analyses are addressed.

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

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

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

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

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

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

3.5.5.3. Acceptability of Continued Operation

This section summarizes the acceptability of continued operation of Bruce B based on the results of the GA and resulting IIP, which is a living record of continuous improvement. The following is addressed:

- Completion of a comprehensive assessment of Bruce B's current organization, governance and processes associated with all aspects of plant operation and the physical plant against the current licencing basis and modern codes and standards;
- Demonstration of the extent to which the Bruce B design, physical plant, operation and applicable governance meet current licencing basis, associated safety goals and fundamental safety principles within the context of defence-in-depth as well as modern codes and standards;
- A well developed state-of-the-art framework which continues to ensure current condition and aging of SSCs important to safety and reliability is understood and effectively managed; and

- An approach that integrates improvements planned or in-progress based on Asset Management and safety basis review inputs with those proposed in the IIP to mitigate SSC aging to ensure continued safe and reliable long-term operation.

Bruce Power's current organizational structure and management system will provide the requisite tools, resources and oversight that will ensure effective execution of the IIP.

3.5.5.4. Document Global Assessment

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

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

3.5.6. Prepare the GA and IIP Report

The results of the steps outlined in Sections 3.5.4.1 and 3.5.5.4 will be documented in a GA and IIP Report to include the following:

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

4. Recording the Output of the PSR

4.1. Database

The successful execution of an ISR or PSR requires the coordination of the efforts of a large number of subject matter experts and special measures to ensure that assessments are performed in a consistent manner, that the assessments are complete and that all findings are tracked to final resolution. A database will be used in the execution of this PSR to ensure that the approach and process are systematic. The database will serve as the single integrated data repository of all PSR activities, and will support the execution of the PSR in the following ways:

- For Safety Factor Reviews:
    - Each code and standard will be uploaded to the database and electronically linked to all the Safety Factors designated to use it in assessments. This facilitates assessments of the same clause from the perspective of different Safety Factors and assessment of only a subset of clauses if that is all that is required;
    - Where necessary, older versions of codes and standards will be uploaded and parsed so that the code-to-code comparisons can be performed. This feature allows for the electronic mapping of the clauses of one version of a standard to that of another version;
    - The database facilitates performing clause-by-clause and high-level assessments;
    - The Review and Comment process on assessments is facilitated by allowing reviewers to enter comments against assessments and capturing their dispositions;
    - The database provides for assigning compliance indicators to each clause and for mapping each gap assessment to a Review Task. This feature facilitates the overall assessment of compliance with the provisions of the Review Task; and
    - All Safety Factor micro-gaps can be mapped to a macro-gap to remove duplication, while maintaining traceability to the original clause gap assessment.
- For Global Assessment:
    - The database enables issues (PIOs) from other sources to be added to the Safety Factor macro-gaps for consolidation into GIOs;
    - The Assessment Framework is imbedded in the database and allows for each GIO to be ranked by selecting the appropriate branch of the value tree and utility assignment parameters;
    - For each GIO the complete definition of the CAs that address it can be entered and mapped to the GI while retaining traceability to the original micro-gaps;

- Similar to the ranking of GIOs, the built-in Assessment Framework allows for the ranking of CAs; and
- The database provides for numerous reports that facilitate the sorting of strengths and gaps to facilitate the global assessment activity.
- The database demonstrates traceability and provides references to the SFRs

- For Integrated Implementation Planning:
  - The database demonstrates traceability and provide references to the GAR
  - The database provides for the definition of the CARDs associated with each corrective action; and
  - The database facilitates the development of the implementation plan for each corrective action.

The quality assurance steps taken during the execution of the Bruce A ISR has been documented, and this will be used during the conduct of the Bruce B PSR for the work performed in the database.

4.2. Contents of the Safety Factor Reports

To facilitate the integration and reporting of the PSR results, a standard template for documenting the reviews will be followed. To ensure consistency in terms of content, level of detail, and presentation of information, the review conducted for each of the Safety Factors will be documented following the Table of Contents given in the following.

Safety Factor Report: Table of Contents

1. Objective and Description

This section is an introduction including the objectives for the Safety Factor and the review tasks. It includes any special circumstances unique to the Bruce B station and an overview of the safety issues.

2. Methodology of Review

This Section records the methodology used for the review. It builds on the Safety Review Tasks described in Appendix A. It should give the reader sufficient information to assess the quality of the review. It includes how the assessment determines compliance with the codes and standards and the evidence to be reviewed.

3. Applicable Codes & Standards

This Section lists the codes and standards in the station's Licensing Basis based on Appendix C and those used as benchmarks for the review. It clarifies which of the codes and standards will be assessed and the type of assessment, based on the guidance given in Section 3.3.3. The codes and standards will be discussed under the appropriate heading as follows:

3.1 Acts and Regulations

For example, the Nuclear Safety and Control Act and Radiation Protection Regulations.

3.2 Power Reactor Operating Licence

Nuclear Power Reactor Operating Licence Bruce Nuclear Generating Stations A and B.

3.3 Regulatory Documents

For example, CNSC REGDOC-2.5.2.

3.4 CSA Standards

This includes the mandatory CSA standards cited in the PROL, as well as non-mandatory standards.

3.5 International Standards

The international standards are primarily the IAEA standards and USA standards as required.

3.6 Other Applicable Standards/Practices

For example, codes and standards not referenced in the licence or listed above.

4. Overview of Applicable Bruce Power Programs and Processes

This section lists the Bruce Power programs, procedures and other station documents in the station's Licensing Basis (Appendix B) as well as relevant lower-tier documents that apply to the Safety Factor and are to be used in the assessments against modern codes, standards and good practices. For each guidance document identified the title and revision of standards it is intended to comply with will also be noted. A discussion of the completeness of the set of documents to be used for the Safety Factor will be included.

5. Results of the Review Tasks

This section reports the results of the review identifying strengths and gaps.

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. This section identifies specific aspects of this Safety Factor that are addressed in, or where more detail is provided in, another Safety Factor Report.

7. Program Assessments and Adequacy of Implementation

This section contains the results of the assessment of relevant programs with reference to self-assessments and audits.

8. Summary and Conclusions

This section provides a summary of the strengths and macro-gaps, as well as an overall conclusion.

9. References

This section lists all the references used in the document and assessments

Appendices

Separate appendices will be provided containing the assessments against codes and standards. This will include the text of clauses assessed, the assessment comments, compliance indicator and references cited.

4.3. Contents of the Global Assessment Report and IIP

4.3.1. GAR Portion of Report

The GAR provides an overall assessment of the safety of the plant for continued operation with an extended operating life based on the integrated results from the Safety Factor Reviews. The content of the GAR portion of the report will include the following recommended by REGDOC-2.3.3:

- summaries of the Safety Factor Reports and identified gaps and strengths;
- overlaps, omissions, and interface issues of the findings from the Safety Factor Reports;
- consolidation of gaps into global issues where appropriate;
- safety significance and risk ranking of all gaps (individual or consolidated as Global Issues);
- corrective actions, safety improvements and appropriate dispositions proposed for all gaps and global issues;
- a global assessment based on the aggregate effect of the findings resulting from all Safety Factor Reports, taking the proposed corrective actions and safety improvements into account, and defence-in-depth; and
- statement of the licensee's assessment of the overall acceptability of operation of the NPP.

4.3.2. IIP Portion of Report

The content of the IIP portion of the report will address the results of the global assessment, and include the following recommended by REGDOC-2.3.3:

- list the corrective actions and safety improvements (including necessary physical NPP modifications) that will address all gaps identified in the PSR, and findings; and
- specify the schedule for implementing the corrective actions and safety improvements.

Each improvement initiative included in the IIP will contain an identifier linking it to the relevant CNSC safety and control area so as to facilitate the CNSC's review.

5. PSR Project Management

Bruce Power will prepare a project management plan to address the PSR project management in accordance with Section 6 of REGDOC-2.3.3 and with BP-PROC-01024, Periodic Safety Reviews [30]. That document will include the project organization (see Figure 5).

The PSR project schedule is shown in Figure 6.

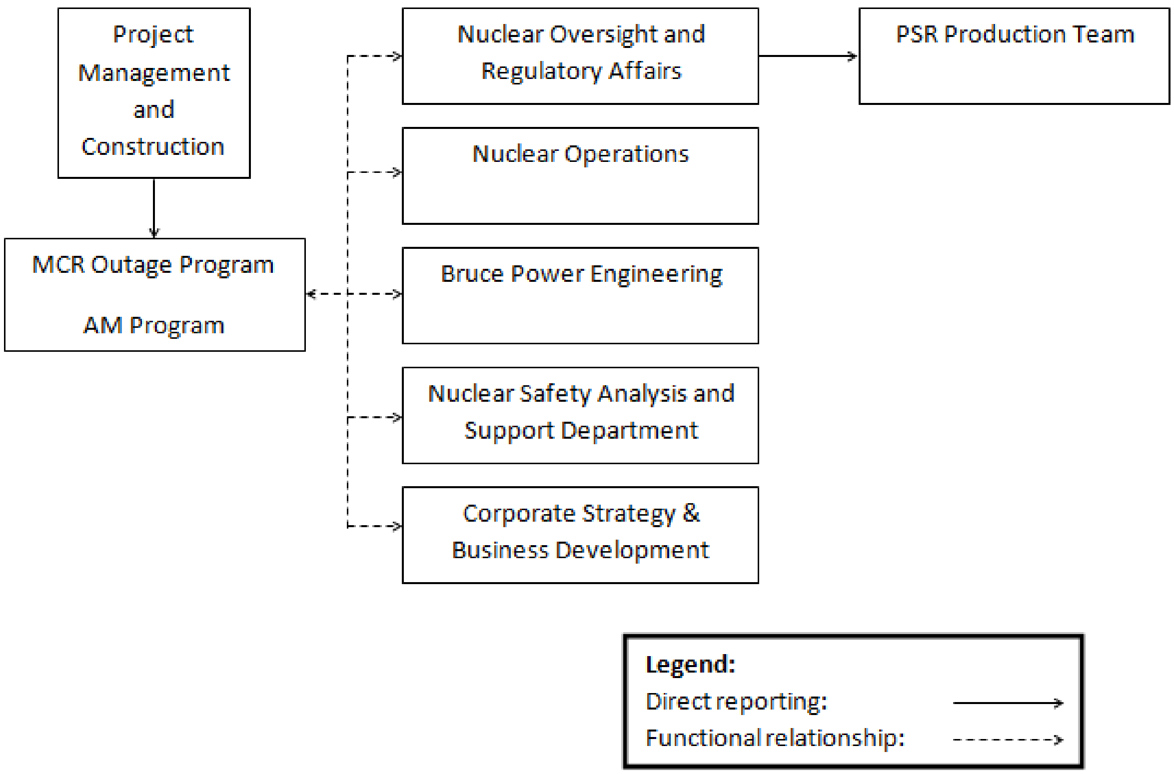


Figure 5: Bruce B PSR Project Organization

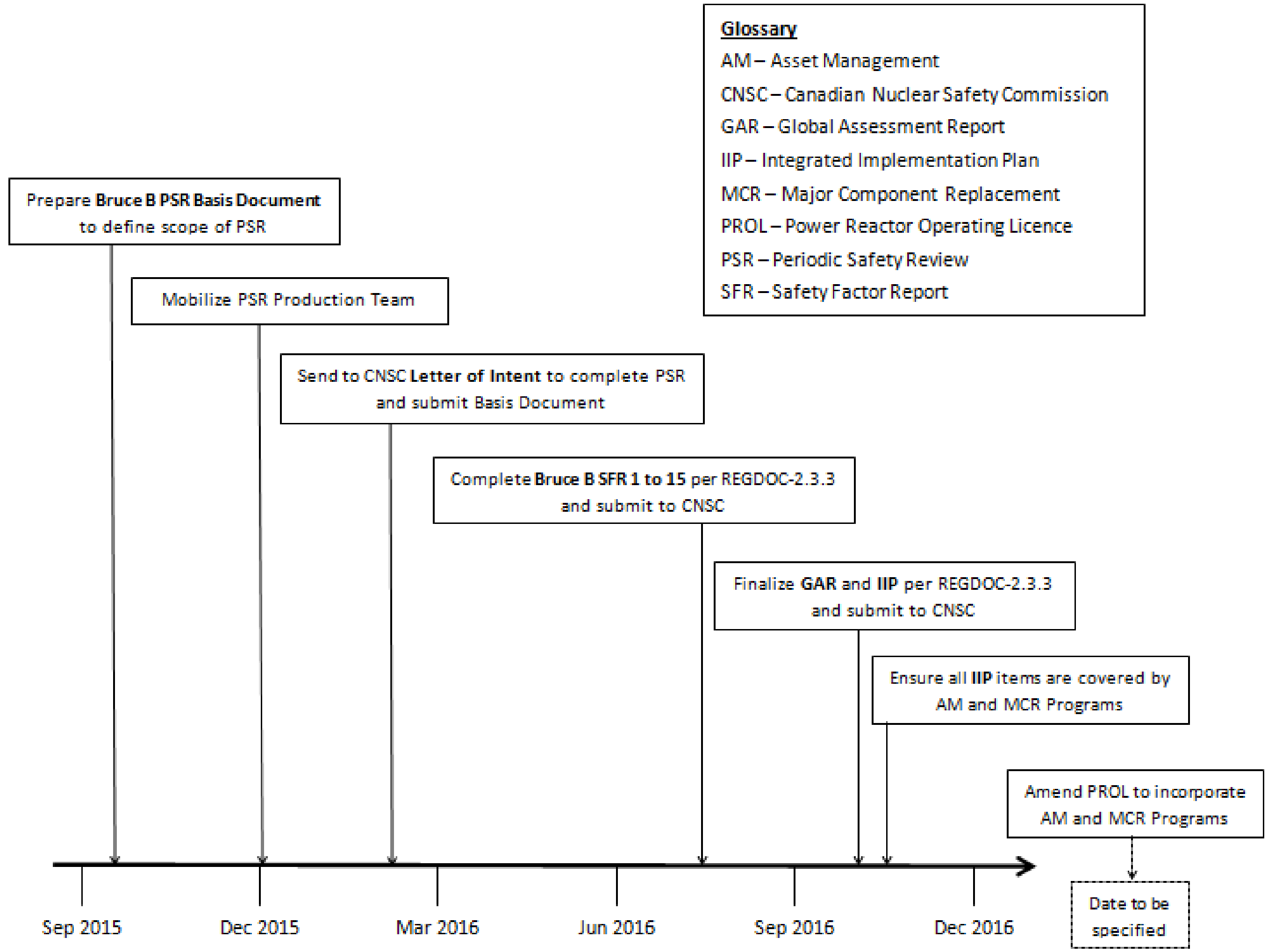


Figure 6: Bruce B PSR Project Schedule

Further aspects of project management are discussed below.

5.1. PSR Project Quality Assurance

PSR work prepared by Bruce Power shall be prepared under the Bruce Power Management System Manual, BP-MSM-1 [31] and CSA N286-05 (Update No. 1), Management System requirements of NPPs, while accordance with BP-PROC-01024, Periodic Safety Reviews [30].

Contractors' quality assurance program shall meet the requirements of ISO 9001. Towards this end, Candesco prepares a Managed Task Plan, which defines the deliverables and activities that they will undertake in support of the project.

5.2. Internal Project Communications

Project progress and issues will be communicated between the PSR Production Team and the PSR Project Management group through weekly progress meetings. In addition, the PSR Production Team will conduct separate weekly internal project meetings. Urgent issues will be communicated through direct contact between the PSR Production Team's project manager and the Bruce Power PSR project manager on an as-needed basis.

Emerging issues that may affect worker or public safety, or that may indicate a non-compliance with the PROL, will be promptly communicated in writing between the PSR Project Manager and the Nuclear Oversight and Regulatory Affairs Vice President.

Further details of internal communications will be provided in the project management plan.

5.3. Communications with the CNSC

All communications with the CNSC will be through Nuclear Oversight and Regulatory Affairs, the single point of contact with the regulator.

5.4. Project Staff Training

All staff directly involved with the production of the Safety Factor Reports, GAR, and IIP will be provided with training consisting, at minimum, of the following topics:

1. Project quality assurance requirements;
2. Project roles and responsibilities;
3. Content of deliverables;
4. Conduct of codes and standards assessments; and
5. Content and use of the database.

Notwithstanding the above-noted training, the Managed Task Plan states that general qualifications required by Candesco project team members are *inter alia* a university degree in science/engineering (all project team members, except Project Administration), with at least 15 years of experience in the nuclear industry for technical leads.

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Safety Factor 2 – Actual Condition of SSCs, Candesco Report, K-421231-00012-R00

Safety Factor 3 – Equipment Qualification, Candesco Report, K-421231-00013-R00

Safety Factor 4 – Ageing, Candesco Report, K-421231-00014-R00

Safety Factor 5 – Deterministic Safety Analysis, Candesco Report, K-421231-00015-R00

Safety Factor 6 – Probabilistic Safety Analysis, Candesco Report, K-421231-00016-R00

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Appendix A – Safety Factor Review Tasks

IAEA SSG-25 [2] and CNSC REGDOC-2.3.3 [3] provide guidance on the scope of the review. SSG-25 breaks down the safety design, operation and management of an NPP into 14 Safety Factors; CNSC REGDOC-2.3.3 provides the elements for assessing Radiation Protection, since SSG-25 does not address Radiation Protection as a separate Safety Factor. Each Safety Factor covers a number of review tasks so that the PSR is a comprehensive safety review.

One of the steps in preparing the PSR Basis is the definition of the scope and intent of the reviews to be performed for each Safety Factor. This Appendix identifies Safety Factor Review Tasks to be performed for the PSR.

The codes and standards that apply for each Safety Factor are given in Appendix C.

The program documents that will be reviewed under each review task are listed in Appendix B.

The following Table of Review Tasks for the Safety Factors is organized as follows by safety area.

A.1 Plant Safety Factors;

A.2 Safety Analysis Safety Factors;

A.3 Safety Performance and Operating Experience (OPEX) Safety Factors;

A.4 Management Safety Factors;

A.5 Environmental Safety Factors; and

A.6 Radiation Protection Safety Factor.

In each Safety Area the Safety Factors are listed with the objective for each factor, followed by the review tasks for each Safety Factor from SSG-25 and REGDOC-2.3.3.

A.1. Plant Safety Factors

SF1 Plant Design – The objective of the review of plant design is to determine the adequacy of the design of the nuclear power plant and its documentation by assessment against modern national and international standards and practices.

The review covers SSCs important to safety unless modified otherwise. The scope of the tasks will depend on the extent of changes in standards and/or the licensing basis since the previous ISRs and PSRs. The review of plant design (including site characteristics) includes the following tasks:

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

SF2 Actual Condition of SSCs – The objective of the review is to determine the actual condition of SSCs important to safety and whether it is adequate for them to meet their design requirements. In addition, the review should confirm that the condition of SSCs is properly documented.

The review of the actual condition of the SSCs important to the safety of the nuclear power plant will include examination of the following aspects for the selected SSCs:

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

SF3 Equipment Qualification – The objective of the review is to determine whether equipment important to safety is qualified to (including for environmental conditions) and whether this qualification is being maintained through an adequate program of maintenance, inspection and testing that provides confidence in the delivery of safety functions.

The review tasks for SF3 are as follows:

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

SF4 Ageing – The objective of the review of ageing is to determine whether ageing aspects affecting SSCs important to safety are being effectively managed and whether an effective ageing management program is in place so that all required safety functions will be delivered for the design lifetime of the plant and, if it is proposed, for long term operation.

1. The following programmatic and technical aspects of the ageing management program will be addressed:
    a. The timely detection and mitigation of ageing mechanisms and/or ageing effects;
    b. The comprehensiveness of the program, i.e., does it address all SSCs important to safety?
    c. The effectiveness of operating and maintenance policies and/or procedures for managing the ageing of replaceable components;
    d. Evaluation and documentation of potential ageing degradation that may affect the safety functions of SSCs important to safety;
    e. Management of the effects of ageing on those parts of the nuclear power plant that will be required for safety when the nuclear reactor has ceased operation, for example the spent fuel storage facilities;
    f. Performance indicators;
    g. Record keeping.

2. The review will address the following technical aspects:
    a. Ageing management methodology;
    b. The operating organization’s understanding of dominant ageing mechanisms and phenomena, including knowledge of actual safety margins;
    c. Availability of data for assessing ageing degradation, including baseline data and operating and maintenance histories;
    d. Acceptance criteria and required safety margins for SSCs important to safety;
    e. Operating guidelines aimed at controlling and/or moderating the rate of ageing degradation;
    f. Methods for monitoring ageing and for mitigation of ageing effects;
    g. Awareness of the physical condition of SSCs important to safety and any features that could limit service life;
    h. Understanding and control of ageing of all materials (including consumables, such as lubricants) and SSCs that could impair their safety functions; and
    i. Obsolescence of technology used in the nuclear power plant.

A.2. Safety Analysis Safety Factors

SF5 Deterministic Safety Analysis – The objective of the review of the deterministic safety analysis is to determine to what extent the existing safety analysis remains valid when the following aspects have been taken into account: actual plant design; the actual condition of SSCs and their predicted state at the end of the period covered by the PSR; current deterministic methods; and current safety standards and knowledge. In addition, the review should also identify any gaps relating to the application of the defense in depth concept.

The review of the deterministic safety analysis will include the following tasks:

1. Review of the application of analytical methods, guidelines and computer codes used in the existing deterministic safety analysis and comparison with current standards and requirements;
2. Review of the current state of the deterministic safety analysis (original analysis and updated analysis) for the completeness of the set of postulated initiating events forming the design basis, with consideration given to feedback of operating experience from plants of a similar design, in Canada;
3. Evaluation of whether the assumptions made in performing the deterministic safety analysis remain valid given the actual condition of the plant;
4. Evaluation of whether the actual operational conditions of the plant meet the acceptance criteria for the design basis;
5. Evaluation of whether the assumptions used in the deterministic safety analysis are in accordance with current regulations and standards;
6. Review of the application of the concept of defence in depth;
7. Evaluation of whether appropriate deterministic methods have been used for development and validation of emergency operating procedures and the accident management program at the plant;
8. Evaluation of whether calculated radiation doses and releases of radioactive material in normal and accident conditions meet regulatory requirements and expectations; and
9. Analysis of the functional adequacy and reliability of systems and components, the impact on safety of internal and external events, equipment failures and human errors, the adequacy and effectiveness of engineering and administrative measures to prevent and mitigate accidents.

SF6 Probabilistic Safety Analysis – The objectives of the review of the PSA are to determine:

- The extent to which the existing 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;
- Whether the existing scope and application of PSA are sufficient.

The review of the PSA will include the following aspects:

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 PSR global assessment, for example, to compare proposed improvement options.

SF7 Hazard Analysis – The objective of the review of hazard analysis is to determine the adequacy of protection of the nuclear power plant against internal and external hazards with account taken of the actual plant design, actual site characteristics, the actual condition of SSCs and their predicted state at the end of the period covered by the PSR, and current analytical methods, safety standards and knowledge.

The review tasks are as follows:

1. For each internal or external hazard identified, include the adequacy of the protection, with account taken of the following:
    a. The credible magnitude and associated frequency of occurrence of the hazard;
    b. Current safety standards;
    c. Current understanding of environmental effects;
    d. The capability of the plant to withstand the hazard as claimed in the safety case, based on its current condition and with allowance given to predicted ageing degradation;
    e. The appropriateness of procedures to cover operator actions claimed to prevent or mitigate the hazard.
2. Check list of internal and external hazards for completeness.

    a. The following is a list of representative external hazards that may affect plant safety (additional site specific external hazards will be included under this Safety Factor if appropriate):

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

    b. The following is a representative list of internal hazards that may affect plant safety (additional site specific internal hazards will be included under this Safety Factor if appropriate):

i. Fire (including measures for prevention, detection and suppression of fire);
ii. Flooding;
iii. Pipe whip;
iv. Missiles and drops of heavy loads;
v. Steam release;
vi. Hot gas release;
vii. Cold gas release;
viii. Deluge and spray;
ix. Explosion;
x. Electromagnetic or radio frequency interference;
xi. Toxic and/or corrosive liquids and gases;
xii. Vibration;
xiii. Subsidence;
xiv. High humidity;
xv. Structural collapse;
xvi. Loss of internal and external services (cooling water, electricity, etc.);
xvii. High voltage transients; and
xviii. Loss or low capacity of air conditioning (which may lead to high temperatures).

A.3. Safety Performance and OPEX and R&D Safety Factors

SF8 Safety Performance – The objective of the review of safety performance is to determine whether the plant's safety performance indicators and records of operating experience, including the evaluation of root causes of plant events, indicate the need for safety improvements.

The review of safety performance will be restricted to review of operating experience at the plant only, as follows:

1. The review of safety performance will evaluate whether the plant has in place appropriate processes for the routine recording and evaluation of safety related operating experience, including:
    a. Safety related incidents, low level events and near misses;
    b. Safety related operational data;
    c. Maintenance, inspection and testing;
    d. Replacements of SSCs important to safety owing to failure or obsolescence;
    e. Modifications, either temporary or permanent, to SSCs important to safety;
    f. Unavailability of safety systems;
    g. Radiation doses (to workers, including contractors);
    h. Off-site contamination and radiation levels;
    i. Discharges of radioactive effluents;
    j. Generation of radioactive waste;
    k. Compliance with regulatory requirements.
2. Where safety performance indicators are used, the review will consider their adequacy and effectiveness, applying trend analysis and comparing performance levels with those for other plants in Canada;
3. The review will consider the effectiveness of the processes and methodology used to evaluate and assess operating experience and trends. The findings of the reviews of other Safety Factors will be taken into account when undertaking this task;
4. Records of radiation doses and radioactive effluents will be reviewed to determine whether these are within prescribed limits, as low as reasonably achievable and adequately managed. Although radiation risks will be considered in all Safety Factors, the review of this Safety Factor will examine specifically data on radiation doses and radioactive effluents and the effectiveness of the radiation protection measures in place. The review will take into account the types of activity being undertaken at the plant, which may not be directly comparable with those at other nuclear power plants in Canada; and
5. Data on the generation of radioactive waste will be reviewed to determine whether operation of the plant is being optimized to minimize the quantities of waste being generated and accumulated, taking into account the national policy on radioactive discharges and international treaties, standards and criteria.

SF9 OPEX and R&D – The objective is to determine whether there is adequate feedback of safety experience from nuclear power plants (both internal and external) and of the findings of research.

The review will identify operating experience reports and other information that may be important to nuclear safety at other plants owned by the operating organization, together with relevant experience and national and international research findings from nuclear and non-nuclear facilities both in Canada and in other States. It will be verified that this information has been properly considered within the plant's routine evaluation processes and that appropriate action has been taken.

The specific review tasks are as follows:

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

A.4. Management Safety Factors

SF10 Organization and Administration – The objective is to determine whether the organization and administration are adequate for the safe operation of the nuclear power plant. The review tasks are as follows:

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

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

    a. A review of the safety policy to verify that it states that safety takes precedence over production and to confirm that this policy is effectively implemented;
    b. A review of procedures to ensure that nuclear and radiation safety are properly controlled and that appropriate measures are applied consistently and conscientiously by all staff;
    c. An assessment of the extent to which a questioning attitude exists and conservative decision making is undertaken in the organization;
    d. Verification that there is a strong drive to ensure that all events that may be instructive are reported and investigated to discover root causes and that timely feedback is provided to appropriate staff on findings and remedial actions;
    e. Verification that unsafe acts and conditions are identified and challenged in a constructive manner wherever and whenever they are encountered by plant employees and external staff (contractors);
    f. Verification that the organization has a learning culture and that it strives continuously for improvements and new ideas, and benchmarks against and searches out best practices and new technologies;
    g. Verification that there is an established and effective process for communication of safety issues;
    h. Verification that there is a process in place for prioritization of safety issues, with realistic objectives and timescales, that ensures that these issues receive proper resources;
    i. Verification that there is a method in place for achieving and maintaining clarity of the organizational structure and managing changes in accountability for matters affecting safety; and
    j. Verification that there is adequate training in safety culture, particularly for managers.

SF11 Procedures - The objective of the review of procedures is to determine whether the operating organization's processes for managing, implementing and adhering to operating and working procedures and for maintaining compliance with operational limits and conditions and regulatory requirements are adequate and effective and ensure plant safety.

The review will examine a selection of the following procedures:

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

SF12 Human Factors – The objective of the review of human factors is to determine the status of the various human factors that may affect the safe operation of the nuclear power plant.

The review tasks are as follows:

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

SF13 Emergency Planning – The objective of the review of emergency planning is to determine whether the operating organization has adequate plans, staff, facilities and equipment for dealing with emergencies and whether the operating organization's arrangements have been adequately coordinated with local and national systems and are regularly exercised.

The review tasks are as follows:

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

A.5. Environment Safety Factor

SF14 Radiological Impact on the Environment – The objective of the review of the radiological impact of the nuclear power plant on the environment is to determine whether the operating organization has an adequate program for surveillance of the radiological impact of the plant on the environment, which ensures that emissions are properly controlled and are as low as reasonably achievable.

The review will include the following:

Verification whether the monitoring program is appropriate and sufficiently comprehensive. In particular, the review should verify that the radiological impact of the plant on the environment is not significant compared with that due to other sources of radiation.

Additionally:

a. Concentrations of radionuclides in air, water (including river water, sea water and groundwater), soil, agricultural and marine products and animals are being monitored by the operating organization or by an independent public organization and are trended, and appropriate corrective actions are taken in the event that action levels are exceeded;
b. Potential new sources of radiological impact have been recognized by the operating organization;
c. Sampling and measurement methods are consistent with current standards;
d. Records of discharges of effluents are being monitored and trended and appropriate actions are taken to remain within established limits and to keep such discharges as low as reasonably achievable;
e. On-site monitoring is undertaken at locations and using methods that have a high probability of the prompt detection of a release of radioactive material to the environment;
f. Off-site monitoring for contamination levels and radiation levels is adequate and corrective actions are taken to keep such levels as low as reasonably achievable;
g. Actions have been taken to clean up contamination where reasonable and practicable;
h. Alarm systems to respond to unplanned releases of radioactive material from on-site facilities are suitably designed and available and will remain available in the future;
i. Appropriate data have been published on the environmental impact of the plant;
j. Changes in the use of areas around the site have been taken into account in the development of monitoring programmes.

A.6. Radiation Protection Safety Factor

SF15 Radiation Protection – The objective of the review of radiation protection is to determine:

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

The review tasks are as follows:

1. Reactor design features for radiation protection;
2. Radiation protection equipment and instrumentation for radiation monitoring;
3. Radiation protection aspects during nuclear emergencies; and
4. Radiation protection operating experience.

Appendix B – Bruce Power Documents Supporting the Bruce B Operating Licence

The Bruce A and B Power Reactor Operating Licence – PROL 18.00/2020 [2] references regulatory documents, codes and standards that apply to the operation of the station. It also lists the key Bruce B station documents that support the operating licence. These Bruce Power documents, along with those listed in the licence renewal application submitted in 2013 [22] in support of the current operating licence, from supplementary submissions in 2014 [23] [24], and from additional documents identified in the LCH as being part of the licensing basis [21] are listed in Table B-1. The CNSC Regulations under the Nuclear Safety and Control Act require extensive information be submitted in support of a licence application. For Bruce B, the information is contained in the Bruce Power letter, "Application for the Renewal of the Power Reactor Operating Licence for Bruce Nuclear Generating Station B" [22] and in the subsequent submissions with supplementary information [23] [24].

The majority of applicable Bruce Power documents are grouped in 10 attachments to the letter of application.

Index of Attachments

Attachment A:	Activity to be Licensed and its Purpose
Attachment B:	Nuclear Substances – Bruce B
Attachment C:	Site Description and Plan – Bruce B
Attachment D:	Land Ownership and Control
Attachment E:	Financial Guarantees – Bruce B
Attachment F:	Evidence of Adequate Nuclear Liability Insurance – Bruce B
Attachment G:	Operational Support Documentation
Attachment H:	CNSC Licences Held By Bruce Power
Attachment I:	Station Improvement Plans
Attachment J:	Status of Open CNSC Action Items

Table B-1: Bruce Power Governance Documents

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
B-CTP-35400-00001	Irradiated Fuel Shipping	R002	N	Y
B-HBK-09500-00003	Training – Performance Objectives and Criteria	R000	N	Y
B-HBK-09500-00005	Training – Performance Objectives and Criteria Evaluator Reference Material	R000	N	Y
NK29-PIP-31100-00001	Bruce Nuclear Generating Station Fuel Channel Periodic Inspection Program	R000	Y	N
B-PLAN-20000-00001	Life Cycle Management Plan for Civil Structures	R000	Y	N
B-PLAN-31100-00001	Fuel Channel Life Cycle Management Plan	R005	Y	N
B-PLAN-33110-00001	Steam Generator and Preheater Life Cycle Management Plan	R004	Y	N
B-REP-09034-00001	Bruce Power Reliability Program	R000	N	Y
B-REP-31100-00003	Fuel Channel Condition Assessment	R003	N	Y

[4] As of December 31, 2015.

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
B-ST-03480-10000	Radionuclide effluent Monitoring System Requirements	R001	N	Y
B-PLAN-33126-00001[5]	Feeder Piping Life Cycle Management Plan	Superseded by B-LCM-33126-00001	Y	Y
BP-ERP-00001	Shift Emergency Controller (SEC)	R018	N	Y
BP-ERP-00042	Emergency Recovery Director	Superseded by BP-ERP-00061-R000 & BP-ERP-00062-R002	N	Y
BP-MSM-1	Management System Manual	R012	Y	Y
BP-MSM-1 Sheet 0001	MSM-Bruce Power Program Matrix	R020	N	Y
BP-NSAS-00016	Integrated Ageing Management for Safety Analysis	R000	N	Y
BP-OPP-00001	Operating Policies and Principles – Bruce B	R015	Y	Y
BP-OPP-00002	Operating Policies and Principles – Bruce A	R013	Y	Y
BP-PLAN-00001	Bruce Power Nuclear Emergency Response Plan	R004	Y	Y

[5] This is the correct document number as seen in the Licence Renewal Application, however it is a typo in the LCH (BP-PLAN-33126-00001).

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PLAN-00003	Bruce Power Electricity Emergency Plan	R005	N	Y
BP-PLAN-00004	Business Continuity Management	R008	N	Y
BP-PLAN-00005	Radioactive Material Transportation Emergency Response Plan	R005	Y	Y
BP-PLAN-00006	Conventional Emergency Plan	R001	Y	Y
BP-PLAN-00008	Fire Safety Management	R003	Y	Y
B-PLAN-07292-00002	Bruce Nuclear Generating Station B – Spill Prevention Contingency	R006	N	Y
B-PLAN-07292-00004	Bruce Power Live Exercise and Spill Drill Planning – 5 Year Plan	R003	N	Y
BP-PROC-00003	Cobalt Handling	R002	Y	Y
BP-PROC-00005	Limits to Hours of Work	R013	Y	N
BP-PROC-00010	Emergency Preparedness Drills And Exercises	R008	N	Y
BP-PROC-00019	Action Tracking	R002	N	Y
BP-PROC-00020	Employee Temporary Assignment Process (PWU and Management)	R010	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00059	Event Response and Reporting	R021	N	Y
BP-PROC-00060	Station Condition Record Process	R026	N	Y
BP-PROC-00062	Processing External and Internal Operating Experience	R014	N	Y
BP-PROC-00064	Formal Correspondence with the CNSC	R009	N	Y
BP-PROC-00076	Management of the Off-Site Radiological Environmental Monitoring Program	R005	N	Y
BP-PROC-00079	MISA Regulation Requirements and Interpretation	R002	N	Y
BP-PROC-00080	Monitoring of Radioactivity in Effluents	R006	N	Y
BP-PROC-00093	Spills to the Environment	R016	N	Y
BP-PROC-00098	Records Management	R014	N	Y
BP-PROC-00099	Conventional Emissions	R008	N	Y
BP-PROC-00127	Radioactive Liquid Emissions Response Procedure	R011	N	Y
BP-PROC-00133	Hazardous Waste Management Requirements	R005	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00137	Focus Area Self-Assessment	R014	N	Y
BP-PROC-00147	Benchmarking and Conference Activities	R015	N	Y
BP-PROC-00150	Notifications Prior to Maintenance of Fire Systems	R002	N	Y
BP-PROC-00151	Job Filling and Reassignment for Society Represented Positions	R009	N	Y
BP-PROC-00158	Removal of Packaging Material Prior to Entering Bruce A Protected Area, Bruce B Protected Area and COS Zone 2, Zone 3 and the Unzoned Areas	R004	N	Y
BP-PROC-00159	Control of Ignition Sources	R007	N	Y
BP-PROC-00163	Non-Licensed Operators Staffing of Days - Based Rotational Positions	R003	N	Y
BP-PROC-00164	Authorized Nuclear Operator and Unit 0 Control Room Operator Staffing Days - Based Rotational Positions	R003	N	Y
BP-PROC-00171	Radiological Emissions Limits and Action Levels	R017	Y	Y
BP-PROC-00174	Training – Administer Training Exemptions	R004	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00175	Training – Prepare a Training Needs Analysis	R003	N	Y
BP-PROC-00176	Training – Administer Remedial Training	R002	N	Y
BP-PROC-00186	Fire Extinguishers	R002	N	Y
BP-PROC-00187	Fire Protection Impairment Control	R015	N	Y
BP-PROC-00188	Radioactive Material Transportation	R002	N	Y
BP-PROC-00189	Control of Transient Material	R011	N	Y
BP-PROC-00196	Conventional Landfill Waste Disposal	R007	N	Y
BP-PROC-00201	Training – Prepare Tests, Field Checkouts and Question & Answer Banks	R003	N	Y
BP-PROC-00202	Training – Administer Tests and Field Checkouts	R002	N	Y
BP-PROC-00203	Training – Prepare a Job Analysis	R002	N	Y
BP-PROC-00204	Training – Perform a Task Analysis	R001	N	Y
BP-PROC-00205	Training – Prepare a Cost Analysis	Obsolete	N	Y
BP-PROC-00206	Training – Prepare Learning Objectives	R001	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00207	Training – Prepare a Job Performance Measure	R002	N	Y
BP-PROC-00208	Training – Prepare Lesson Plans, Course Materials and Training Aids	R002	N	Y
BP-PROC-00209	Training - Administer Training Change Control	R007	N	Y
BP-PROC-00210	Training – Administer Vendor Training	R001	N	Y
BP-PROC-00211	Training – Administer On-the-Job Training (OJT) and On-the-Job Evaluation (OJE)	R003	N	Y
BP-PROC-00212	Training – Administer Training Delivery	R004	N	Y
BP-PROC-00213	Training – Administer Training Evaluation	R004	N	Y
BP-PROC-00214	Training – Administer TIMS	R003	N	Y
BP-PROC-00215	Training- Administer Training Scheduling	R008	N	Y
BP-PROC-00216	Training – Prepare a Training and Qualification Description (TQD)	R006	N	Y
BP-PROC-00217	M&TE Calibration Program Requirements	R010	N	Y
BP-PROC-00219	Source Segregation and Recycling Program	R007	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00222	On-Boarding of Managers	R005	N	Y
BP-PROC-00238	Retention Process for Bruce Power Records	R011	N	Y
BP-PROC-00259	Fire Protection for Relocatable Structures	R002	N	Y
BP-PROC-00262	Warehouse Operations	R008	N	Y
BP-PROC-00268	Safety Related System Testing (SST) Program	R004	N	Y
BP-PROC-00271	Observation and Coaching Procedure	R004	N	Y
BP-PROC-00279	Joint Health and Safety Committee Operations	R004	N	Y
BP-PROC-00280	Dosimetry Requirements	R008	Y	N
BP-PROC-00289	Fire Hose and Couplings	R001	N	Y
BP-PROC-00299	Training – Administer Staff Capability	R003	N	Y
BP-PROC-00301	Reactivity Management	R002	N	Y
BP-PROC-00306	Chemical Risk Assessment Procedure	R011	N	Y
BP-PROC-00307	Control of Handling, Storage and Shipping	R014	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00317	Crisis Management	R004	N	Y
BP-PROC-00324	Nuclear Criticality Safety Management	R004	Y	N
BP-PROC-00328	New Work Prioritization and Approval	R013	N	Y
BP-PROC-00329	On-Line Work Management Process	R014	N	Y
BP-PROC-00334	Periodic Inspection	R003	N	Y
BP-PROC-00335	Design Management	R006	N	Y
BP-PROC-00342	Planned Outage Management	R005	N	Y
BP-PROC-00342 Sheet 1	Planned Outage Preparation Milestones	R008	N	Y
BP-PROC-00342 Sheet 2	Scope Review Panel	R010	N	Y
BP-PROC-00342 Sheet 3	Planned Outage and Preparation	R008	N	Y
BP-PROC-00342 Sheet 4	Planned Outage Schedule Development	R003	N	Y
BP-PROC-00342 Sheet 5	Planned Outage Execution	R010	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00342 Sheet 6	Planned Outage Close Out	R008	N	Y
BP-PROC-00342 Sheet 7	Outage Control Centre	R008	N	Y
BP-PROC-00342 Sheet 8	Managing and Executing Work During Outage	R003	N	Y
BP-PROC-00343	Forced Outage Management	R005	N	Y
BP-PROC-00344	Five Year Outage Plan	R000	N	Y
BP-PROC-00360	Training – Administer Critical Knowledge Retention	R001	N	Y
BP-PROC-00363	Nuclear Safety Assessment	R003	N	Y
BP-PROC-00389	Conventional Safety Programs	R002	N	Y
BP-PROC-00400	Life Cycle Management of Critical SSCs	R002	N	Y
BP-PROC-00412	Trending, Analyzing, and Reporting of SCRs	R005	N	Y
BP-PROC-00452	Core Management	R000	N	Y
BP-PROC-00455	Fuel Procurement	R001	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00460	Fuel Handling	R002	N	Y
BP-PROC-00465	Hiring Process (Regular Positions)	R009	N	Y
BP-PROC-00468	Workforce Planning Process	R007	N	Y
BP-PROC-00496	Trouble Shooting Plant Equipment	R005	N	Y
BP-PROC-00506	Effectiveness Reviews	R007	N	Y
BP-PROC-00510	Certification Training – Job Analysis for Certification Training Programs	R004	N	Y
BP-PROC-00511	Certification Training – Task Analysis for Certification Training Program	R004	N	Y
BP-PROC-00512	Certification Training – Training Design for Certification Training Programs	R003	N	Y
BP-PROC-00513	Certification Training – Training Development for Certification Training Programs	R004	N	Y
BP-PROC-00517	Health Physics Response to Radiation Overexposure	R001	N	Y
BP-PROC-00518	Root Cause Investigation	R006	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00535	Written Communication with Environmental Regulators	R001	N	Y
BP-PROC-00539	Design Change Package	R015	N	Y
BP-PROC-00543	Task Planning	R011	N	Y
BP-PROC-00565	Certification Training – Independence and Confidentiality Requirements for Development and Implementation of Initial Certification and Re-Certification Examinations	R011	N	Y
BP-PROC-00566	Certification Training – Standards and Methodology for Certification Training Process	R008	N	Y
BP-PROC-00567	Certification Training – Development and Administration of Diagnostic Simulator-Based Re-Certification	R013	N	Y
BP-PROC-00568	Certification Training – Development and Administration of Comprehensive Written and Oral Examinations for Initial Training Programs	R012	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00569	Certification Training – Development and Administration of Comprehensive Simulator-Based Examinations for Initial Certification Training Programs	R011	N	Y
BP-PROC-00570	Certification Training – Development and Administration of Written Re-Certification Examinations and Examination Material for Certified Staff	R005	N	Y
BP-PROC-00571	Certification Training – Development and Administration of Comprehensive Simulator-Based Re-Certification Examinations (CST) for Certified Staff	R012	N	Y
BP-PROC-00572	Certification Training – Remedial Training for Certification Training Programs	R006	N	Y
BP-PROC-00574	Certification Training – Filing and Retention of Certification Training Records	R007	N	Y
BP-PROC-00575	Mentored Structured Learning	R002	N	Y
BP-PROC-00576	Certification Training – Conduct of Continuing Training and Re-Certification Testing	R006	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00577	Certification Training – Conduct of Initial Certification Training	R002	N	Y
BP-PROC-00583	Ministry of Labour Interface Management	R001	N	Y
BP-PROC-00589	Outage Fundamentals	R003	N	Y
BP-PROC-00595	Training Fundamentals	R001	N	Y
BP-PROC-00596	Occupational Health and Safety Hazards and Applicable Legal Requirements	R004	N	Y
BP-PROC-00604	Training – Non-Licensed Operator Continuing Training	R006	N	Y
BP-PROC-00610	Fitness for Duty	R002	Y	N
BP-PROC-00617	Human Performance Tools for Workers	R005	N	Y
BP-PROC-00619	Occupational Health and Safety Management Review	R002	N	Y
BP-PROC-00651	Safety Performance Metrics and Monitoring	R001	N	Y
BP-PROC-00653	Training – Administer Continuing Training	R003	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00659	Severe Accident Management Procedure	R002	N	Y
BP-PROC-00667	Certification Training – Conduct of Continuing Training for Authorized/Responsible Health Physicists	R000	N	Y
BP-PROC-00695	Maintenance Program Basis	R003	N	Y
BP-PROC-00696	Maintenance Organization	R002	N	Y
BP-PROC-00697	Maintenance Activities	R001	N	Y
BP-PROC-00698	Structures, Systems or Components (SSC) Monitoring	R001	N	Y
BP-PROC-00699	Maintenance Work	R003	N	Y
BP-PROC-00703	Change Management Guidance	R001	Y	N
BP-PROC-00707	Conventional Safety Instrumentation Management	R003	N	Y
BP-PROC-00722	Pandemic Response	R003	N	Y
BP-PROC-00734	Plant Status Control	R005	N	Y
BP-PROC-00752	Training – Prepare Computer Based Training	R003	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00773	Hazardous Waste Management and Disposal Requirements	R001	N	Y
BP-PROC-00778	Scoping and Identification of Critical SSCs	R001	N	Y
BP-PROC-00779	Continuing Equipment Reliability Improvement	R000	N	Y
BP-PROC-00780	PM Implementation	R001	N	Y
BP-PROC-00781	Performance Monitoring	R002	N	Y
BP-PROC-00782	Problem Identification and Resolution	R000	N	Y
BP-PROC-00783	Long-Term Planning and Life Cycle Management	R001	N	Y
BP-PROC-00784	Cybersecurity	R001	Y	N
BP-PROC-00794	Monitoring Human Performance	R002	N	Y
BP-PROC-00795	Human Performance Tools for Knowledge Workers	R000	N	Y
BP-PROC-00811	Procedure Alterations	R002	N	Y
BP-PROC-00815	Visual Inspection of Containment Boundary Components	R002	Y	N

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROC-00325	Buried Piping Inspection Program	R004	N	Y
BP-PROC-00833	Reporting to the CNSC	R000	N	Y
BP-PROC-00839	Reporting to the CNSC/IAEA Safeguards	R000	N	Y
BP-PROC-00842	Compressed Gas Storage	R002	N	Y
BP-PROC-00848	Training – Administer Manual Credits and Credit Corrections	R00	N	Y
BP-PROC-00857	Fire Barriers	R000	N	Y
BP-PROC-00870	Spill Management and Contaminated Lands Program	R002	N	Y
BP-PROC-00872	Conventional Safety Observations and Inspections	Cancelled	N	Y
BP-PROC-00878	Radioactive Waste Management	R000	N	Y
BP-PROC-00888	Conventional and Hazardous Waste Management	R000	N	Y
BP-PROC-00919	Stakeholder Information Disclosure	R000	N	Y
BP-PROC-00928	Conventional Emissions Air	R000	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROG-00.02	Environmental Safety Management	R008	Y	Y
BP-PROG-00.04	Pressure Boundary Quality Assurance Program	R020	Y	Y
BP-PROG-00.06	Health and Safety Management	R008	Y	Y
BP-PROG-00.07	Human Performance Program	R010	Y	Y
BP-PROG-01.02	Bruce Power Management System (BPMS) Management	R007	Y	N
BP-PROG-01.06	Operating Experience Program	R014	Y	Y
BP-PROG-01.07	Corrective Action	R010	Y	Y
BP-PROG-02.01	Worker Staffing	R009	Y	Y
BP-PROG-02.02	Worker Learning and Qualification	R013	Y	Y
BP-PROG-02.04	Worker Development and Performance Management	R010	Y	N
BP-PROG-03.01	Document Management	R015	Y	Y
BP-PROG-05.01	Supply Chain	R013	Y	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROG-05.03	Site Services	R009	N	Y
BP-PROG-06.03	CNSC Interface Management	R003	Y	Y
BP-PROG-08.01	Emergency Management Program	R007 Title of R007 changed to "Emergency Measures Program"	Y	Y
BP-PROG-08.02	Nuclear Security	R006	Y	Y
BP-PROG-09.02	Stakeholder Interaction	R005	Y	Y
BP-PROG-10.01	Plant Design Basis Management	R008	Y	Y
BP-PROG-10.02	Engineering Change Control	R009	Y	Y
BP-PROG-10.03	Configuration Management	R005	Y	N
BP-PROG-11.01	Equipment Reliability	R004	Y	Y
BP-PROG-11.02	On-Line Work Management Program	R006	Y	Y
BP-PROG-11.03	Outage Work Management	R005	Y	Y
BP-PROG-11.04	Plant Maintenance	R006	Y	Y
BP-PROG-12.01	Conduct of Plant Operations	R007	Y	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-PROG-12.02	Chemistry Management	R005	Y	Y
BP-PROG-12.03	Fuel Management Program	R003	Y	Y
BP-PROG-12.05	Radiation Protection Program	R003	Y	Y
BP-PROG-12.06	Radioactive Waste Management	Replaced by BP-PROC-00878, Radioactive Waste Management	N	Y
BP-PROG-12.07	Heavy Water Management	R001	Y	N
BP-PROG-14.01	Project Management and Construction	R005	Y	N
BP-PROG-14.02	Contractor Management	R005	Y	N
BP-PROG-15.01	Nuclear Oversight Management	R004	Y	Y
BP-RPP-00009	Dose Limits and Exposure Control	R008	Y	N
BP-RPP-00044	ALARA Program	R003	Y	N
BP-SM-00010	Chemical Storage Cabinets	R013	N	Y
BP-SM-00045	Bruce Power Asbestos Control	R006	N	Y
BP-SM-00054	Hazardous Materials Approval Process	Superseded by BP-PROC-00306	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
BP-SM-00075	Transportation of Dangerous Goods Non-radioactive	R002	N	Y
BP-SM-00080	Workplace Hazardous Materials Information System	R001	N	Y
B-ST-07290-10000	MISA Compliance Technical Specifications	R000	N	Y
DIV-EM-00007	Emergency Measure Program Assessment	Superseded by SEC-EPP-00007	N	Y
DIV-ENG-00004	Engineering Evaluations	R008	N	Y
DIV-ENG-00010	Probabilistic Risk Assessment Process	R009	N	Y
DIV-ENG-00017	System and Item Classification	R001	Y	N
DIV-ENG-00018	Design Registration and Reconciliation	R001	Y	N
DIV-OPA-00001	Station Shift Complement – Bruce A	R010	Y	N
DIV-OPB-00001	Station Shift Complement – Bruce B	R006	Y	Y
DIV-OPA-00002	Bruce A Role Descriptions for Licence-Related Positions	R002	Y	N
DIV-OPB-00002	Bruce B Role Descriptions for Licence-Related Positions	R001	Y	N

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
DIV-OPA-00002	Appendix B: Shift Manager Appendix B – Control Room Shift Supervisor Appendix B: Authorized Nuclear Operator Appendix B: Unit 0 control Room Operator	R002	Y	Y
DOM-BBOP-00001	Bruce B Operations – Division Organization Manual	R002	N	Y
DOM-CA-00001	Corporate Affairs	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-CSBD-00001	Corporate Strategy and Business Development Division Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-DMES-00001	Engineering Support- Division Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-DMSE-00001	Station Engineering – Division Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-FS-00001	Financial – Division Organizational Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-FINCON-00001	Corporate Controller-Division Organizational Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-IT-00001	IT Infrastructures and Operations	Superseded by BP-MSM-1 Sheet 0002	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
DOM-LAW-00001	Law Division Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-MKT-00001	Power Marketing	R010	N	Y
DOM-NORA-00001	Nuclear Oversight and Regulatory Affairs	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-NUCOS-00001	Nuclear Operations Support	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-OPA-00001	Bruce B Station – Division Organization Manual	R002	N	Y
DOM-OPB-00001	Bruce B Station – Division Organization Manual	R004	N	Y
DOM-OTG-00002	Outage Division – Division Organization Manual	R002	N	Y
DOM-OTGB-00001	Bruce B Outage	R001	N	Y
DOM-PMCBMG-00001	PMC Business Management	R001	N	Y
DOM-PMCCON-00001	PMC Construction Division Organization Manual	Obsolete	N	Y
DOM-PMCPP-00001	Programs and Projects Division	Superseded by BP-MSM-1 Sheet 0002	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
DOM-PFM-00001	Portfolio Management – Division Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-SC-00001	Supply Chain – Division Organizational Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-SSER-00001	Site Services	Superseded by BP-MSM-1 Sheet 0002	N	Y
DOM-TRG-00001	Training – Division Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
DPT-CHM-00003	Control of System Chemistry	R006	N	Y
DPT-CHM-00006	Analytical Capability	R012	N	Y
DPT-ERO-00008	Emergency Services Team Drills And Exercises	R004	N	Y
DPT-NSAS-00002	Safety Report Analysis Update Process Overview	R004	N	Y
DPT-NSAS-00003	Guidelines for Evaluating and Prioritizing Safety Report Issues	R004	N	Y
GOM-CNOB-00001	Nuclear Operations Bruce B – Group Organization Manual	R000	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
GOM-HR-00001	Human Resources Group	Superseded by BP-MSM-1 Sheet 0002	N	Y
GOM-FIN-00001	Finance and Commercial Services Group Organization Manual	Superseded by BP-MSM-1 Sheet 0002	N	Y
GOM-NMS-00001	Nuclear Maintenance Services	Superseded by BP-MSM-1 Sheet 0002	N	Y
GOM-PMC-00001	Project Management and Construction Group Organization	Obsolete	N	Y
GRP-OPS-00003	Certification Training – Copilot Procedure	R014	N	Y
GRP-OPS-00025	Expectations of Duty Managers	R005	N	Y
GRP-OPS-00038	Bruce A and Bruce B Operations Standards and Expectations	R009	N	Y
GRP-OPS-00055	Fitness for Duty Considerations for Shift Complement Staff Held Over for More than 13 Hours	R001	N	Y
NK21/29-OM-35030	Nuclear Fuel Location and Storage History	R001/R018	N	Y
NK21/29-OM-35100	New Fuel Transfer and Storage	R027/R026	N	Y
NK29-OM-35310/35320	Irradiated Fuel Transfer and Auxiliaries	R034	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
NK21/29-OM-35390 Section 6.0/4.6	Bruce Used Fuel Dry Storage	R000/R0026	N	Y
NK21/29-OM-35370 LCH gives -35390 as the SCI for Bruce B, whereas licence application gives -35370	Safeguards Operating Manual	R003/R002	Y	Y
NK29-OM-79500	Chemical Waste Management (Bruce B)	R006	N	Y
NK21/29-PIP-03641.2-00001	Bruce A/B Periodic Inspection Program for Unit 1/5	R004	Y	N
NK21/29-PIP-03641.2-00002	Bruce A/B Periodic Inspection Program for Unit 2/6	R004	Y	N
NK21/29-PIP-03641.2-00003	Bruce A/B Periodic Inspection Program for Unit 3/7	R004	Y	N
NK21/29-PIP-03641.2-00004	Bruce A/B Periodic Inspection Program for Unit 4/8	R004	Y	N
NK21-PIP-03642-00001	Bruce A Periodic Inspection Program for Unit 0 and 1 to 4 Containment Components	R001	Y	N
NK29-PIP-03642-00001	Bruce A Periodic Inspection Program for Unit 0 and 5 to 8 Containment Components	R002	Y	N

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
NK21/29-PIP-21100-00001	CSA N287.7-08 – Periodic Inspection Program for Bruce NGS A/B Concrete Containment Structures and Appurtenances (Excluding Vacuum Building)	R002	Y	N
NK21/29-PIP-25100-00001	CSA N287.7-08 – Periodic Inspection Program for Bruce NGS B Vacuum Building	R001	Y	N
NK21/29-PLAN-03480-00001	Site Emission Monitoring Plans	R001	N	Y
NK21/29-REP-03482-00002/3	Derived Release Limits and Actions Levels for Bruce Nuclear Generating Station A/B	R003	Y	Y
NK29-SR-01320-00001	Bruce B Safety Report, Part 1: Plant and Site Description	R005	Y	Y
NK29-SR-01320-00002	Bruce B Safety Report, Part 2: Plant Components and Systems	R005	Y	Y
NK29-SR-01320-00003	Bruce B Safety Report, Part 3: Accident Analysis	R004	Y	Y
NK37-DRAW-10200-10001	Site Facilities Plan of the Bruce Nuclear Power Development Lots 11 to 28 and Part of 29 and 30	R000	Y	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
NK37-SMP-79500-00001	Chemical Waste Transfer	R002	N	Y
NK37-SMP-79500-00002	Waste Oil Handling and Transfer	R001	N	Y
OM-WC-79500	Waste Chemical Transfer Facility (non-rad materials only)	R002	N	Y
SEC-CHD-00001	Guidelines for Preparing/Revising Chemistry Specifications	R001	N	Y
SEC-DOCM-00035	Records Retrieval and Secure Storage	R014	N	Y
SEC-PE-00014	Selection of Shipping Storage and Handling Requirements	R004	N	Y
SEC-RPR-00022	Action Levels	R003	N	Y
SEC-RPR-00040	Responsibilities of an Authorized Health Physicist	R003	Y	N
SEC-SIMM-00001	Simulator Validation	R001	N	Y
SEC-SIMM-00002	Simulator Change Control	R001	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
SEC-SSPE-00002	Maintenance of Air Conditioning and Refrigeration Equipment To Reduce Emissions of Refrigerants	R005	N	Y
TQD-00009	Engineering Support Personnel	R008	N	Y
TQD-00012	Bruce A and Bruce B Authorized Nuclear Operator Initial Training	R005	N	Y
TQD-00013	Bruce A and Bruce B Control Room Shift Supervisor/Shift Manager	R007	N	Y
TQD-00014	Certified Staff Continuing Training and Recertification- Testing	R006	N	Y
TQD-00015	Bruce A and Bruce B Certified Unit 0 Control Room Operator Initial	R005	N	Y
TQD-00019	Non-Licensed Operators – Generating Units	R005	N	Y
TQD-00022	Control Maintenance	R007	N	Y
TQD-00023	Mechanical Maintenance	R005	N	Y
TQD-00030	Non-Licensed Operators Unit 0	R006	N	Y
TQD-00031	Non-Licensed Operators – General Services	R000	N	Y

Document Number	Document Title	Current Revision[4]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application or Supplemental Information (Y/N)
TQD-00032	Non-Licensed Operators – Fuel Handling	R005	N	Y
TQD-00036	Chemical Technician	R008	N	Y
TQD-00046	Radiation Protection Technician	R002	N	Y
TQD-00075	Health Physicist	R003	N	Y

Appendix C – Regulatory Documents, Guides, Codes and Standards in the Licence and to be Assessed in the PSR

Table C-1 identifies the codes, standards and guides that are relevant to this PSR. In doing so, it identifies:

a) the codes and standards that are referenced in the current PROL [7] [22] [23] [24] or LCH [21];

b) the latest revisions of the codes and standards that have been assessed in previous safety reviews, along with the date of the assessment;

c) the current revision of the codes and standards. The year (or, in some cases, year/month) is identified in the column;

d) whether a new assessment is required, and if so, the type of assessment. The cells are shaded for codes or standards for which a new assessment is required; and

e) the relevant Safety Factors.

Modern revisions of some of the codes and standards listed in Table C-1 have been identified in the licence renewal application and supplementary submissions for the current PROL. Reference [24] identifies if there are transition plans for such codes and standards. Such transition plans will be used to address gaps in cases where a code or standard is included in the current licence, as done in the Bruce A ISR.

Table C-1: List of Codes and Standards

Legend	
NA	Not assessed
CBC	FULL Clause by Clause
PCBC	Partial CBC
HL	High Level
CV	Confirm Validity of Previous Assessments (Section 3 only)
CTC/HL	Code to code was performed generally with a HL assessment
CTC/CBC	Code to code was performed with a clause by clause assessment of new or different clauses
2SF	Refers to assessment performed in another SFR
PROL	In PROL & no further assessment needed for review tasks
PROL-T	In PROL, with transition plan for compliance

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
ANSI/HPS N13.1-1999	1999	Sampling and Monitoring Releases of Airborne Radioactive Substances From the Stacks and Duct of Nuclear Facilities														HL		
ANSI/NIRMA CM 1.0-2007	2007	Guidelines for Configuration Management of Nuclear Facilities	HL															

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
ASME BPVC Section III	2015	Rules for Construction of Nuclear Power Plant Components	HL															
ASME BPVC Section VIII	2015	Design and Fabrication of Pressure Vessels	HL															
ASME B31.1	2014	Code for Power Piping	HL															
CNSC EG-1	2005	Requirements and Guidelines for Written and Oral Certification Examinations for Shift Personnel at Nuclear Power Plants								NA		NA		NA				PROL
CNSC EG-2	2004	Requirements and Guidelines for Simulator-Based Certification Examinations for Shift Personnel at Nuclear Power Plants								NA		NA		NA				PROL
CNSC G-129	Rev 1 (2004/10)	Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'														HL	2SF	
CNSC G-144	2006/05	Trip Parameter Acceptance					HL											
CNSC G-149	2000/10	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	HL				HL											
CNSC G-228	2001/03	Developing and Using Action Levels														HL	HL	
CNSC G-276	2003/06	Human Factors Engineering Program Plans	HL											HL				

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CNSC G-278	2003/06	Human Factors Verification and Validation Plan												NA				PROL
CNSC G-323	2007/08	Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement										2SF		HL				
CNSC Internal Guidance 2009/05	2009	Requirements for the Requalification Testing of Certified Shift Personnel at Nuclear Power Plants								NA		NA		NA				PROL
CNSC Internal Guidance 2010/08	2010	CNSC Expectations for Licensee Hours of Work Limits - Objectives and Criteria								NA		NA		NA				
CNSC P-325	2006/05	Nuclear Emergency Management													CV			
CNSC R-10	1977/01	The Use of Two Shutdown Systems in Reactors	NA				NA			NA								
CNSC R-77	1987/10	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	NA				CV		CV									
CNSC R-116	1995/01	Requirements for Leak Testing Selected Sealed Radiation Sources															NA	
CNSC RD/GD-99.3	2012/03	Public Information and Disclosure								NA	NA		NA		NA	NA		PROL
CNSC RD-204	2008/02	Certification of Persons Working at Nuclear Power Plants								NA		NA		NA				PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CNSC RD/GD-210	2012/11	Maintenance Programs for Nuclear Power Plants			NA	NA				NA		NA		NA				PROL
CNSC RD-327	2010/12	Nuclear Criticality Safety	NA															PROL-T
CNSC RD-346	2008/11	Site Evaluation for New Nuclear Power Plants	NA						CV							CV		
CNSC REGDOC-1.6.1[6]	2015/10	Licence Application Guide: Nuclear Substances and Radiation Devices															NA	Licence[7]
CNSC REGDOC-2.10.1	2014/10	Nuclear Emergency Preparedness and Response													CBC			PROL
CNSC REGDOC-2.2.2	2014/08	Personnel Training								2SF		2SF		CBC				
CNSC REGDOC-2.3.2	2015/09[8]	Accident Management Severe Accident Management Programs for Nuclear Reactors					PCBC								CBC			PROL

[6] Supersedes CNSC RD/GD 371

[7] The CNSC Nuclear Substances & Radiation Devices licence authorizes the licensee to possess transfer, import, export, use and store nuclear substances and prescribed equipment for the purposes of “industrial radiography” “throughout the Bruce Power site”. It is assumed that any activities performed using or in support of the nuclear substances and prescribed equipment possessed under this licence are performed in accordance with the terms and conditions of this licence and CNSC REGDOC-1.6.1. Thus, CNSC REGDOC-1.6.1 will not be assessed as part of this PSR.

[8] Version 2013/09 is listed in the PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CNSC REGDOC-2.3.3	2015/04[9]	Periodic Safety Reviews	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	PROL
CNSC REGDOC-2.4.1	2014/05	Safety Analysis For Nuclear Power Plants (Current: Deterministic Safety Analysis)					CBC		PCBC									PROL-T
CNSC REGDOC-2.4.2	2014/05	Probabilistic Safety Assessment For Nuclear Power Plants	2SF					CBC										PROL-T
CNSC REGDOC-2.5.2	2014/05	Design of Reactor Facilities: Nuclear Power Plants	CBC			PCBC	PCBC	PCBC	PCBC					2SF				
CNSC REGDOC-2.6.3	2014/03	Fitness for Service: Ageing Management				NA												PROL-T
CNSC REGDOC-2.9.1	2013/09	Environmental Protection, Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills											2SF			CTC/HL		PROL-T
CNSC REGDOC-3.1.1	2014/05	Reporting Requirements for Operating Nuclear Power Plants				NA				NA	NA	NA	NA			NA	NA	PROL
CNSC RD/GD-98	2012/06	Reliability programs for Nuclear Power Plants		NA		NA		NA										PROL

[9] Version 2014 is listed in the PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CNSC S-106	2006/05	Technical and Quality Assurance Requirements for Dosimetry Services															NA	
CSA B51	2014	Boiler, Pressure Vessel, and Pressure Piping Code	CV	CV														
CSA N1600	2014	General Requirements for Nuclear Emergency Management Programs													HL			
CSA N285.0	2012 Update 1 (2013/09) Update 2 (2014/11)	General Requirements For Pressure-Retaining Systems And Components In CANDU Nuclear Power Plants	NA															PROL-T
CSA N285.4	2014[10]	Periodic Inspection of CANDU Nuclear Power Plant Components		NA		NA												PROL-T
CSA N285.5	2013[11]	Periodic Inspection of CANDU Nuclear Power Plant Containment Components				CTC/HL												PROL
CSA N285.8	2015	Technical Requirements for In-service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors				HL												

[10] Version 2009 Update 2 (2011/06) is listed in the PROL

[11] Version 2008 is listed in the PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CSA N286.7-99	1999 (R2012)	Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants	NA	NA			NA	NA	NA									PROL
CSA N286-05	2012	Management System Requirements for Nuclear Power Plants	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	PROL-T
CSA N287.1	2014	General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	CTC/HL			CTC/CBC												
CSA N287.2	2008 (R2013)	Material Requirements for Concrete Containment structures in CANDU nuclear power plants	CV															
CSA N287.3	2014	Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	HL															
CSA N287.4	2009 (R2014)	Construction, Fabrication, and installation requirements for Concrete Containment Structures for CANDU nuclear power plants	CV															
CSA N287.5	2011	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	CV															
CSA N287.6	2011	Pre-operational proof and leakage rate testing requirements for concrete containment structures for nuclear power plants	NA															
CSA N287.7	2008 with Update 1 (2010) (R2013)	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants				NA												PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CSA N288.1	2014[12]	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities														HL		PROL
CSA N288.2	2014	Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors					HL								2SF			
CSA N288.3.4	2013	Performance testing of nuclear air cleaning systems at nuclear facilities								HL						HL		
CSA N288.4	2010 (R2015)	Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	HL													HL		PROL-T
CSA N288.5	2011	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills														HL		PROL-T
CSA N288.6	2012	Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills														HL		PROL-T
CSA N289.1	2008 (R2013)	General requirements for seismic design and qualification of CANDU nuclear power plants	HL		HL													
CSA N289.2	2010 (R2015)	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	HL		HL													

[12] Version 2008 with Update 1 2011 is listed in the PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CSA N289.3	2010 (R2015)	Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants	HL		HL													
CSA N289.4	2012	Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants	HL		HL													
CSA N289.5	2012	Seismic Instrumentation Requirements for CANDU Nuclear Power Plants	HL		HL													
CSA N290.0-11	2011	General Requirements for safety systems of nuclear power plants	HL															
CSA N290.1	2013	Requirements for the Shutdown Systems of CANDU Nuclear Power Plants	CBC				PCBC											
CSA N290.2	2011	Requirements for emergency core cooling systems of nuclear plants	HL															
CSA N290.3	2011	Requirements for the containment system of nuclear plants	HL															
CSA N290.4	2011	Requirements for Reactor Control Systems of Nuclear Power Plants	CV				CV											
CSA N290.5	2006 (R2011)	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	CV				CV											
CSA N290.6	2009 (R2014)	Requirements for monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	CV				CV											

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
CSA N290.11	2013	Requirements for reactor heat removal capability during outage of nuclear power plants	HL															
CSA N290.12	2014	Human Factors in Design for Nuclear Power Plants	2SF											CBC				
CSA N290.13	2005[13] (R2015)	Environmental Qualification of Equipment for CANDU Nuclear Power Plants	NA	NA	NA	NA	NA		NA									PROL
CSA N290.15	2010 (R2015)	Requirements for the safe operating envelope of nuclear power plants	NA	NA			NA	NA	NA	NA								PROL-T
CSA N291	2015/01	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	HL	HL		PCBC												
CSA N292.3	2014	Management of Low- and Intermediate-Level Radioactive Waste											CBC			2SF		
CSA N293	2012	Fire Protection For CANDU Nuclear Power Plants	2SF						CTC/CBC									PROL-T
CSA Z731	2003 (R2014)	General requirements for nuclear emergency management programs													NA			CV

[13] Version R2010 is listed in the PROL

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
Darlington DG-38-03650-1	-	Purpose and Application of Nuclear Safety in Design	NA															
Darlington DG-38-03650-2A	-	Common Mode Incidents – Overview and Design Requirements	NA						NA									
Darlington DG-38-03650-2B	-	Common Mode Incidents – Seismic Design	NA						NA									
Darlington DG-38-03650-3	-	Limiting Consequential Damage of Postulated Pipe Ruptures	NA						NA									
Darlington DG-38-03650-4	-	Shutdown Systems	NA															
Darlington DG-38-03650-5	-	Emergency Coolant Injection	NA															
Darlington DG-38-03650-6	-	Containment	NA															
Darlington DG-38-03650-7	-	Extensions of the Containment Envelope	NA															
Darlington DG-38-03650-8	-	Environmental Qualification of Safety Related Equipment	NA		NA													
Darlington DG-38-03650-9	-	Safety Assessments	NA															
IAEA NS-G-3.2	2002/03	Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants														HL	NA	

Document No.	Current Version	Title	Assignment Type															Rationale (as necessary)
			SF1	SF2	SF3	SF4	SF5	SF6	SF7	SF8	SF9	SF10	SF11	SF12	SF13	SF14	SF15	
IAEA SSG-25	2013/03	Periodic Safety Review for Nuclear Power Plants	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	NA	
IAEA SSR-2/2	2011/07	Safety of Nuclear Power Plants: Commissioning and Operation Specific Safety Requirements										2SF	CBC					
INPO 91-014	Rev 1 (1995/10)	Guidelines for Radiological Protection at Nuclear Power Stations															NA	
NBC	2010 First Revision and Errata (2012/12)	National Building Code of Canada	NA															
NFC	2010	National Fire Code	NA															
NFPA-805	2015	Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants	HL															
NUREG-0700	2002	Human System Interface Design Review Guidelines												HL				
WANO GL 2004-01	2004	Guidelines for Radiological Protection at Nuclear Power Stations															CBC	

Appendix D – Global Assessment Framework

D.1. Introduction

The objective of developing an assessment framework is to devise a systematic methodology and establish a common basis for assessing the relative importance of addressing global issues in terms of aspects such as their safety significance. The same framework is also used to assess the importance of practicable improvements and associated corrective actions for the development of the IIP.

The Global Assessment Framework can be used for ranking and prioritization to answer questions such as the following:

- How should gaps be consolidated into GIOs?
- Which GIOs are the most important?
- How should the GIOs be addressed?
- Which GIOs should be addressed first?

These questions are interrelated, multi-facetted, and sometimes involve competing objectives. Moreover, the outcomes of potential answers to some of these questions are uncertain. An overarching set of values, principles, or goals is needed that can guide these activities and that would “drive” the whole process in a comprehensive, systematic and consistent manner through all the steps to develop an integrated and coordinated set of improvement initiatives.

More specifically, a process is needed to decide on the importance ranking and prioritization of the issues and potential improvements identified through the PSR and other assessment activities. This requires a multi-objective, multi-attribute decision support model to be formulated as follows:

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

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

The resulting prioritization and ranking framework is embedded in the database discussed in Section 4.1. The Value Tree will have three tiers below the cardinal objective. The first two tiers are utilized in the development, ranking and prioritization of Global Issues. The third tier is utilized in the development, ranking and prioritization of corrective actions to address Global Issues.

D.2. Structure of the Value Tree

The Value Tree and the pairwise comparisons for each branch are developed by the Integrated Implementation Plan Project Team (IIPPT), which is a multi-disciplinary team that has been involved in the full application of the PSR with specific expertise in CANDU design, operation, inspection and maintenance, safety analysis, licensing and management of the same.

D.3. Main Branches of Value Tree

In structuring any decision problem it is important to determine exactly what the cardinal objective is and what sub-objectives need to be considered to support it in order to determine the fundamental dimensions of the values to make decisions. A useful technique to structure those values is to make use of a Value Tree. A Value Tree begins with a cardinal objective and a set of fundamental objectives as its main branches. Each fundamental objective is then expanded and supported with more specific objectives. A systematic comparison and assessment of these sub-objectives establishes how each is valued in achieving the cardinal objective.

D.3.1. Definition of Cardinal Objective

The two cardinal objectives in the long-term operation of Bruce B are well known and support Bruce Power's value of 'Safety First' and key result areas of Nuclear Performance Index, Safety Performance and Commercial Index. They are stated as follows:

- Enhanced confidence in the continued safety of Bruce B; and
- Enhanced confidence in the reliability of electricity production by Bruce B for an extended life.

Since these two cardinal objectives are mutually supportive and not in conflict, they can be combined into a single value statement, as follows:

> **Enhanced confidence in the continued safety of Bruce B and reliability of electricity production for an extended life**

This cardinal objective is also referred to as the Tier 0 Objective.

D.3.2. Main Branches of the Value Tree

Having defined “Enhanced confidence in the continued safety of Bruce B and reliability of electricity production for an extended life” as the cardinal objective of the Value Tree, the fundamental supporting objectives can be formulated in terms of the main branches of the Value Tree, also called “Tier 1 Branches”, as shown in Figure D-1.

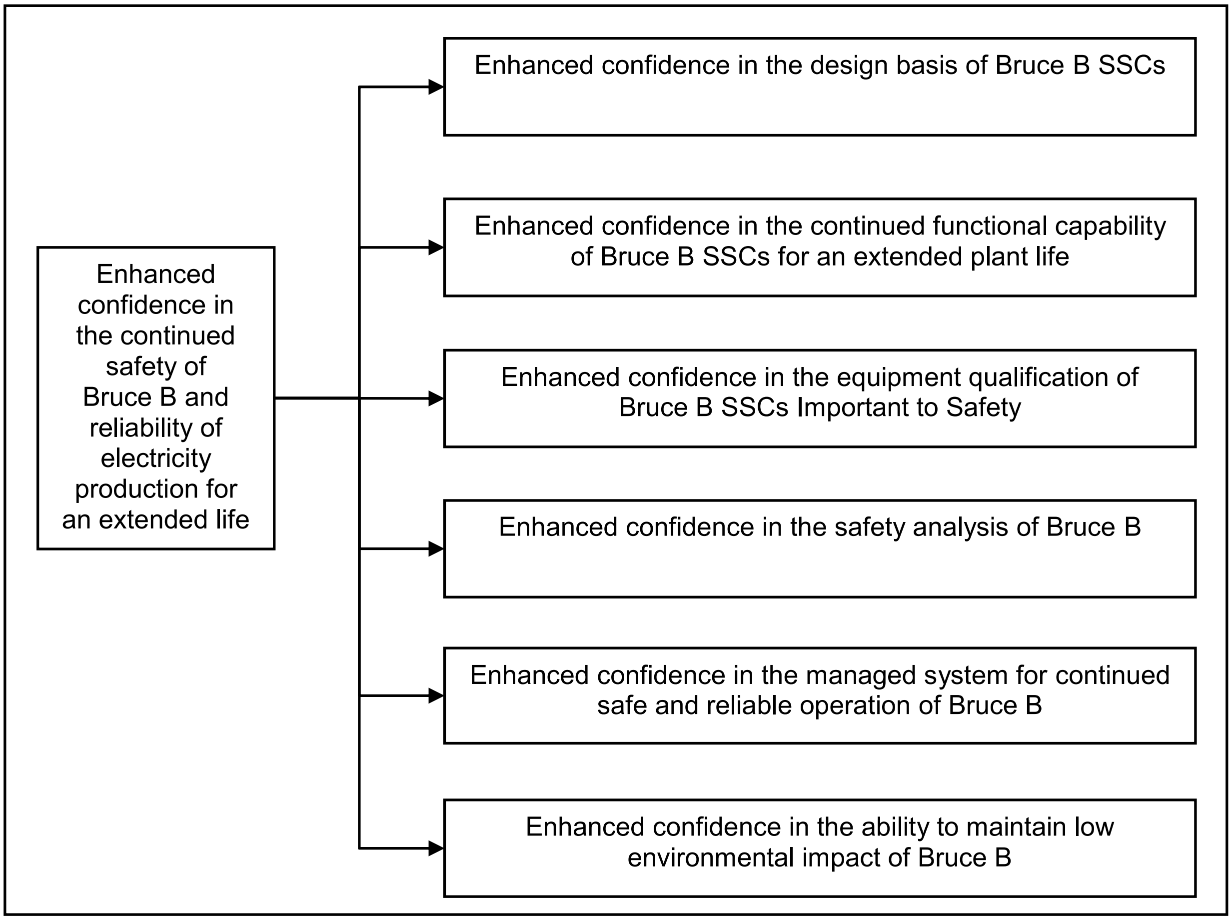

Figure D-1: Tier 1 Branches

The basis for these branches is recognition that Bruce B, in compliance with its Power Reactor Operating Licence (PROL) and associated regulatory framework, must:

- Continue to conform with its design basis;
- Be operated well to achieve safety and reliable electricity production in accordance with its design in accordance with its managed system;
- Have an adequate safety and hazard analysis to demonstrate the facility’s safety; and
- Achieve adequate environmental performance.

Meeting Canada's international obligations, i.e., safeguards, is not a branch of the Value Tree because safeguards do not contribute to the cardinal objective. This subject is treated outside of this methodology, as is security. Improvement opportunities and initiatives associated with safeguards and security are also excluded from the assessment. This means that any safeguards or security related improvement initiatives or commitments in-place will be implemented in accordance with Bruce Power's obligations under the law.

Since one of the key goals of the PSR is to assess whether enhancements should be made to better align with modern standards, these key elements are considered for further subdivision at the second level of the tree. It is important to note that no issue is advantaged or disadvantaged in terms of its rank by the extent of subdivision of the Value Tree. Each Tier 1 branch of the Value Tree is discussed below, together with an overall view of its current state that needs to be taken into consideration for pair wise comparisons.

D.3.2.1. Enhanced Confidence in the Design Basis of Bruce B SSCs

The design branch primarily considers maintenance of the current design basis and potential for improvements to the current design basis, as would be expected in modern standards, given the age of Bruce B and considering the safety improvements implemented since the plant was put into operation.

The original design basis of Bruce B cannot meet all provisions of the applicable modern design codes and standards. A significant level of design improvement to meet modern codes and standards for a new NPP would be required to enhance the design basis to a level comparable to those required for new NPPs. These improvements would require fundamental changes to the SSCs in place, most of which are impracticable to implement, as they would require systemic changes affecting the plant SSCs as a whole. However, since the beginning of Bruce B operation, safety upgrades and supplementary design and safety analyses have been continually implemented to comply with those provisions of the PROL that required design upgrades with high priority and as an integral part of continued safe and reliable operation. A recent example is the implementation of practicable design changes in response to the Fukushima Action Items. It should also be noted that assessments of the design against modern versions of the original design requirements since the plant was put into operation have shown that the original design of Bruce A, including the safety upgrades incorporated, provide an acceptable safety basis at all levels of defence-in-depth, and it is anticipated that a similar result will be found for Bruce B when the PSR is conducted.

D.3.2.2. Enhanced Confidence in the Continued Functional Capability of Bruce B SSCs for an Extended Plant Life

This branch straddles design and operation, and is related to the confidence in the continued functional capability of Bruce B SSCs to meet their current design and operating requirements through monitoring, surveillance, testing, inspection and maintenance of SSCs in accordance with their design and operating envelope for an extended plant life. This branch is fundamentally different than the previous branch and deals with the understanding of the current condition of Bruce B SSCs, ensuring their continued functional capability and maintaining them in this state. The understanding of condition and ensuring functional capability of SSCs links the plant design basis and safe operation, as well as contributing to continued reliable electricity production for an extended life.

Confidence in the continued functional capability of Bruce B SSCs forms the basis of safe and reliable operation for an extended life. Given the age of Bruce B and the considerations for extended plant life, eliminating gaps and enhancing confidence in the understanding of the current condition of Bruce B SSCs, taking necessary actions to ensure their continued functional capability and maintaining them in this state continues to be the most important aspect of safe and reliable operation. Enhanced confidence in the continued functional capability of the as-built plant is the pillar of event-free operation.

D.3.2.3. Enhanced Confidence in the Equipment Qualification of Bruce B SSCs Important to Safety

This branch straddles design and operation, and relates to equipment qualification, which invokes both design and programmatic elements. It is a separate Tier 1 branch because it was not always an explicit requirement originally considered in the design in a comprehensive and systematic manner as defined in modern codes and standards, at least not with respect to some internal and external hazards and certain accident conditions. This branch is also related to the safety analysis element, as the robustness of the design due to initiating events is demonstrated via hazard and safety analysis.

The original design basis of Bruce B cannot not meet all provisions of the applicable modern design codes and standards with respect to equipment qualification. A significant level of design improvement to meet modern codes and standards may be required to enhance the confidence in the design basis for equipment qualification as compared to a new NPP. Assessments performed to date have shown that the original design of Bruce A, including the safety upgrades (e.g., environmental and seismic qualification) incorporated since the plant was put into operation, provide an acceptable safety basis that is in compliance with the PROL at all levels of defence-in-depth, and it is anticipated that a similar result will be found for Bruce B when the PSR is conducted.

It should be noted that all three of the branches discussed so far address both plant safety and reliability considerations. The first two consider all Bruce B SSCs. The equipment qualification branch relates specifically to SSCs important to safety that are required to operate under accident conditions, as well as hazardous conditions due to internal or external events.

D.3.2.4. Enhanced Confidence in the Safety analysis of Bruce B

The safety analysis is maintained as a single branch in the first level of the Value Tree, primarily because the components of safety analysis (deterministic, probabilistic, hazard analysis) are all just different types of safety analysis that should fit within an overall integrated safety analysis framework. These components are recognized in lower tier branches.

A robust safety case relies on results of safety analyses based on a systematic and comprehensive set of postulated initiating events, state-of-the-art analysis methodologies based on up-to-date experimental data and OPEX and input data that reflects the actual plant that is executed in accordance with procedures that meet applicable quality assurance requirements.

Since Bruce B was put into operation, hazard assessments and the scope of safety analysis have become a progressively more important area as a result of international OPEX and the resulting changes to requirements for systems important to safety, as well as the development of the state-of-the-art analysis methodologies based on new experimental data and OPEX. Recently, the update of safety analysis to address plant ageing and to meet new Canadian regulatory documents has also increased the importance of this area.

D.3.2.5. Enhanced Confidence in the Managed System for Continued Safe and Reliable Operation of Bruce B

The managed system branch is similarly kept to a single branch in the first level of the Value Tree. While there are many facets to the managed system that support safe and reliable operation, they are all so closely inter-related that it would be necessary to include numerous branches to differentiate them at the level of the first branch of the Value Tree, which would result in unnecessary complexity at this level. Such differentiation is established more appropriately in the Tier 2 and Tier 3 branches.

Bruce Power has a mature and robust managed system that is built around the principle of ‘safety first’ as the overarching objective. The managed system has built-in continuous improvement features. Bruce Power’s managed system meets all the regulatory requirements in the PROL associated with management of the plant operations.

D.3.2.6. Enhanced Confidence in the Ability to Maintain Low Environmental Impact of Bruce B

The environmental branch is also kept to a single branch in the first level. Differentiation between normal, accident and post-accident conditions takes place at the Tier 2 branch.

Bruce Power has maintained good environmental performance over the years and its management continues to place higher expectations regarding excellence in environmental performance as an integral aspect of ‘safety first’ principle. Regulatory and public expectations with respect to maintaining a progressively lower environmental impact and improving environmental performance against the regulatory targets continue to be a prominent topic. In addition, the recent OPEX from Fukushima resulted in the implementation of initiatives that are designed to minimize potential environmental impacts of severe accidents.

D.4. Expanding the Value Tree Structure – Tier 2 and Tier 3 Branches

Each of the Tier 1 objectives is expanded into specific supporting objectives at Tier 2, and similarly each Tier 2 objective is expanded into more specific supporting objectives at Tier 3. The resulting Value Tree is shown Table D-1 in full, together with a description of each objective.

Safety Factors have been mapped to the Value Tree as shown in the second last column of Table D-1. This mapping demonstrates that all gaps and findings that will be identified through the PSR process and the associated improvement opportunities can be evaluated in terms of priority and ranking within the framework of the Value Tree as part of the Global Assessment and Integrated Implementation Plan development.

The last column of Table D-1 indicates those levels of defence-in-depth supported by each of the Tier 1, Tier 2 and Tier 3 sub-objectives.

Table D-1: Expanded Value Tree Objectives

T1 #	Tier 1 Contribution to Cardinal Values / Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 / Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 / Tier 3 Description	Applicable SFs	DID Level at Tier 2
1	Enhanced confidence in the design basis of Bruce B SSCs	1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1.1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	1 & 3	1,2,3
		1.2	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	1.2.1	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	1, 2, 3 & 4	1,2,3
		1.3	Enhanced confidence that the design of SSCs meets modern standards	1.3.1	Enhanced confidence that the design of the plant meets the enhanced or new design features and provisions included in the modern codes and standards	1 & 3	1,2,3
				1.3.2	Enhanced confidence that the design analysis/qualification of the plant SSCs meet the enhanced or new analytical/qualification requirements included in the modern codes and standards	1 & 3	
2	Enhanced confidence in the continued functional capability of Bruce B SSCs for an extended plant life	2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).	2.1.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) for an extended plant life	2 & 4	1,2,3,4
				2.1.2	Enhanced confidence in knowledge about the current condition of SSCs Important to Reliability (SIR) for an extended plant life	2 & 4	
		2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended	2.2.1	Enhanced confidence in restoring SIS to a state that achieves the intended functionality and extended plant life	2 & 4	1,2,3,4

T1 #	Tier 1 Contribution to Cardinal Values / Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 / Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 / Tier 3 Description	Applicable SFs	DID Level at Tier 2
			plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment Reliability Program.	2.2.2	Enhanced confidence in restoring SIR to a state that achieves the intended functionality and extended plant life	2 & 4	
		2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and extended plant life. This is achieved through the Maintenance Program activities.	2.3.1	Enhanced confidence in maintaining SIS in a state that achieves reliable operation and safety performance and extended plant life	2 & 4	1,2,3,4
				2.3.2	Enhanced confidence in maintaining SIR in a state that achieves reliable operation and safety performance and extended plant life	2 & 4	
		2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.	2.4.1	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life	2 & 4	1,2,3,4
3	Enhanced confidence in the equipment qualification of Bruce B Systems Important to Safety	3.1	Enhanced confidence in the design specification and implementation of equipment qualification conditions for Systems Important to Safety	3.1.1	Enhanced confidence in the current environmental qualification requirements of SIS resulting from deterministic safety analysis of Design Basis Accidents (DBAs)	1, 3 & 5	1,2,3,4
				3.1.2	Enhanced confidence in the equipment qualification requirements resulting from hazards analysis of internal and external events	1, 3, 6 & 7	
				3.1.3	Enhanced confidence in the equipment qualification conditions and requirements for SSCs not already specified in safety analysis or hazards analysis (e.g. Severe Accident (SA) conditions)	1, 3, 6 & 7	1,2,3,4
4	Enhanced confidence in the safety analysis of Bruce B	4.1	Enhanced confidence in the comprehensiveness of the safety analysis	4.1.1	Enhanced confidence in the completeness of all of the requisite elements of analysis in the current accident analyses included in the current analysis of record	5, 6 & 7	1,2,3,4

T1 #	Tier 1 Contribution to Cardinal Values / Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 / Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 / Tier 3 Description	Applicable SFs	DID Level at Tier 2
				4.1.2	Enhanced confidence in the definition of initiating events and combinations thereof in the current analysis of record	5, 6 & 7	
				4.1.3	Enhanced confidence in the coverage of all initiating events and combinations thereof of the current safety analysis of record	5, 6 & 7	
		4.2	Enhanced confidence in conformance with the applicable safety analysis methods and associated acceptance criteria	4.2.1	Enhanced confidence in the degree to which software used for accident analysis has been validated	5, 6 & 7	1,2,3,4
				4.2.2	Enhanced confidence in the degree to which acceptance criteria used in safety analysis is supported by experimental or operational data	5, 6 & 7	
				4.2.3	Enhanced confidence in the application of modern methodologies and criteria in the conduct of safety analysis	5, 6 & 7	
5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce B	5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience	5.1.1	Enhanced confidence in the selection and training of staff	10 & 12	1,2,3,4
				5.1.2	Enhanced confidence in the dissemination and assimilation of internal and external operating experience	10 & 12	
		5.2	Enhanced confidence in the effectiveness of technical and administrative documentation and interfaces for operators, maintainers and operations support staff	5.2.1	Enhanced confidence in the comprehensiveness and effectiveness of procedures	11	1,2,3,4,5
				5.2.2	Enhanced confidence in the appropriateness, validity and timeliness of plant and process information	1, 2, 3, 5, 6, 10, 11 & 12	
				5.2.3	Enhanced confidence in the appropriateness of plant control interfaces (human factors)	12	
		5.3	Enhanced confidence in a safe work environment	5.3.1	Enhanced confidence in radiation protection	15	1,2,3,4,5

T1 #	Tier 1 Contribution to Cardinal Values / Tier 1 Description	T2 #	Tier 2 Contribution to Tier 1 / Tier 2 Description	T3 #	Tier 3 Contribution to Tier 2 / Tier 3 Description	Applicable SFs	DID Level at Tier 2
				5.3.2	Enhanced confidence in conventional health and safety	8	
		5.4	Enhanced confidence in organizational effectiveness by establishing a management structure and processes that can demonstrate continuous improvement of safe plant operation and safety culture	5.4.1	Enhanced confidence in management system structure and processes	10 & 11	1,2,3,4,5
				5.4.2	Enhanced confidence in safety culture	10, 11 & 12	
				5.4.3	Enhanced confidence in performance monitoring and corrective action	8, 9, 10, 11 & 12	
6	Enhanced confidence in the ability to maintain low environmental impact of Bruce B	6.1	Enhanced confidence in maintaining a low environmental impact during normal operations	6.1.1	Enhanced confidence in low impact of radioactive releases	14 & 15	1,2,3,4,5
				6.1.2	Enhanced confidence in low impact of non-radiological releases	14	
		6.2	Enhanced confidence in the ability to mitigate releases associated with external/internal events	6.2.1	Enhanced confidence in the ability to mitigate releases associated with anticipated operational occurrences and design basis events	13 & 14	1,2,3,4,5
				6.2.2	Enhanced confidence in the ability to mitigate releases associated with beyond design basis events	13	

D.5. Assigning Weights to Objectives in the Value Tree

To account for differences in importance between the objectives that comprise the Value Tree, relative weights are assigned to each objective. The method used for assigning the weights is based on the well known AHP ("The Analytic Hierarchy Process", T. L. Saaty, McGraw Hill Inc, 1980) and comprises the following:

- Use of pairwise comparisons to rank all of the objectives attached to the same branch in terms of importance on a scale from 1 to 9. If there are 'n' objectives attached to the same branch, the result is an n x n reciprocal matrix.
- Compute the eigenvalues of the matrix and find the eigenvector corresponding to the largest eigenvalue λ_{max}.
- Normalize this eigenvector and decompose into its n components.
- Assign the components of the normalized eigenvector as weights to the corresponding objectives on the Value Tree.

- Compute the consistency index (CI) as CI = (λ_{max} - n) / (n – 1).
- Compute the Consistency Ratio (CR) as the ratio of the Consistency Index (CI) for a particular set of judgments to the random index (RI) for a matrix of the same size, as published in ("The Analytic Hierarchy Process", T. L. Saaty, McGraw Hill Inc, 1980). If the CR is less than 10%, the judgment results are considered acceptable.

In practice, this process is approximated by calculating the average of the row entries of the reciprocal matrix after the columns are normalized. The Integrated Implementation Plan Database Tool (IIPDT) is an integrated tool based on Microsoft Access technology that provides traceability of all corrective actions to individual PIOs through groupings into GIOs down to the original source of the issue, i.e., gaps or initiatives considered relevant. The Value Tree and weighting system is also built into the tool and the tool takes care of all ranking calculations.

To compensate for the fact that not all branches of the Value Tree have the same number of tiers, the IIPDT also performs a second normalization of the all weights to make them truly relative and comparable. The identical process was carried out for the Bruce A ISR, and the results of that weight assignment process up to Tier 2 are presented in Table D-2.

Table D-2: Assignment of Weights up to Tier 1 to Objectives in Value Tree for Bruce A

Tier 1			Tier 2			
#	**Tier 1 Description**	**Tier 1 Weight *(A)***	**#**	**Tier 2 Description**	**Tier 2 Weight *(B)***	**Ideal Mode Branch Weight (Note 1)**
1	Enhanced confidence in the design basis of Bruce A SSCs	0.0350	1.1	Enhanced confidence that the design of SSCs is in accordance with the applicable regulations, codes and standards and is accurately described in the design documentation	0.4615	0.0350
			1.2	Enhanced confidence that the actual configuration of the SSCs meet the requirements described in the design documentation	0.4615	0.0350
			1.3	Enhanced confidence that the design of SSCs meets modern standards	0.0769	0.0058
2	Enhanced confidence in the continued functional capability of Bruce A SSCs for an extended plant life	0.4916	2.1	Enhanced confidence in knowledge about the current condition of SSCs Important to Safety (SIS) and Important to Reliability (SIR) for an extended plant life. This includes activities such as SSC health monitoring and reporting, condition assessments, Technical Basis Assessments (TBA) Life Cycle Management Plans (LCMP).	0.0559	0.0570
			2.2	Enhanced confidence in restoring SSCs to a state that achieves the intended functionality and extended plant life. This includes activities such as SSC testing, surveillance and inspections as required by the Equipment Reliability Program.	0.3158	0.3216
			2.3	Enhanced confidence in maintaining SSCs in a state that achieves reliable operation and safety performance and	0.4827	0.4916

Tier 1			Tier 2			
#	**Tier 1 Description**	**Tier 1 Weight *(A)***	**#**	**Tier 2 Description**	**Tier 2 Weight *(B)***	**Ideal Mode Branch Weight (Note 1)**
				extended plant life. This is achieved through the Maintenance Program activities.		
			2.4	Enhanced confidence in operation within the appropriate safe operating envelope for an extended plant life. This is achieved through integration of the OSRs in the plant operating documentation such as OP&Ps, OMs, AIMs, SSTs.	0.1455	0.1482
3	Enhanced confidence in the equipment qualification of Bruce A Systems Important to Safety	0.1475	3.1	Enhanced confidence in the design specification and implementation of equipment qualification conditions for Systems Important to Safety	1.0000	0.1475
4	Enhanced confidence in the safety analysis of Bruce A	0.1401	4.1	Enhanced confidence in the comprehensiveness of the safety analysis	0.8000	0.1401
			4.2	Enhanced confidence in conformance with the applicable safety analysis methods and associated acceptance criteria	0.2000	0.0350
5	Enhanced confidence in the managed system for continued safe and reliable operation of Bruce A	0.0457	5.1	Enhanced confidence in staff capabilities through selection of staff with the right capabilities, training of staff to perform their tasks effectively in accordance with the jurisdictional requirements and continuous learning from internal and external operating experience	0.0745	0.0087
			5.2	Enhanced confidence in the effectiveness of technical and administrative documentation and interfaces for operators, maintainers and operations support staff	0.3939	0.0457
			5.3	Enhanced confidence in a safe work environment	0.3747	0.0435
			5.4	Enhanced confidence in organizational effectiveness by establishing a management structure and processes that can demonstrate continuous improvement of safe plant operation and safety culture	0.1569	0.0182
6	Enhanced confidence in the ability to maintain low environmental impact of Bruce A	0.1401	6.1	Enhanced confidence in maintaining a low environmental impact during normal operations	0.8333	0.1401
			6.2	Enhanced confidence in the ability to mitigate releases associated with external/internal events	0.1667	0.0280

Note:

1. Ideal Mode Branch Weight for Tier 2 = $(A \times B) / B_{max}$

   Ideal Mode Branch Weight for Tier 3 = $(((A \times B) / B_{max}) \times C) / C_{max}$

D.6. Expressing Preferences

The second step of developing a complete decision support system is the development of a measure or rate to judge the relative impact of resolving an issue on the objective that it is associated with. A preference rating system using two attributes is used:

- The **Time** attribute measures the impact of resolving an issue by answering the following question: "If one could somehow correct Issue X immediately (i.e., the corrective action happens overnight), how long would it take to see the assessed gap disappear in relation to the corresponding objective?"
- The **Impact** attribute measures how directly or strongly the issue impacts the objective by asking: "If one could somehow correct Issue X immediately (i.e., the corrective action happens overnight), how direct or big would the impact be on the improvement in the objective?"

For both attributes a rating system on a scale of 1 to 5 was developed as shown in Table D-3 and Table D-4.

Table D-3: Scoring System for the Time Attribute

Time Rating	Definition
1	Resolving the issue will take at least 10 years to have its effect on the objective
2	Resolving the issue will take at least 8 to 10 years to have its effect on the objective
3	Resolving the issue will take at least 6 to 10 years to have its effect on the objective
4	Resolving the issue will take at least 4 to 6 years to have its effect on the objective
5	Resolving the issue will take up to 3 years to have its effect on the objective

Table D-4: Scoring System for the Impact Attribute

Impact Rating	Definition
1	Resolving the issue will have an indirect and negligible impact on the objective
2	Resolving the issue will have an indirect and minor impact on the objective
3	Resolving the issue will have an direct and minor impact on the objective
4	Resolving the issue will have an indirect and major impact on the objective
5	Resolving the issue will have an direct and major impact on the objective

To assist assessors in assigning Impact ratings, the terms in Table D-4 are defined as indicated in the matrix shown in Table D-5. This matrix also relates the INSAG-10 levels of defence-in-depth and the nature of corrective actions needed to resolve an issue to the type of Impact.

The resolution attributes (i.e., Direct, Indirect, Major, Minor) are considered at the level of the Value Tree at which the issue is mapped and within the context of the four rows of Table D-5:

1. Maintaining or Improving the Design Basis;
2. Improving Operational or Safety Performance;
3. Reduction of Uncertainty in the design and operational bases of the Physical Plant and its Operation; and
4. Improvement of the Managed System (enablers) to achieve safe operation and reliable electricity production.

Each improvement is also evaluated within the context of four columns in Table D-5 to establish their contribution as barriers to achieve safe and reliable, i.e., event free operation:

1. New barriers and practices;
2. Augmentation (Recovery) of the Current Barriers and Practices;
3. Improvement (Effectiveness) of the Current Barriers and Practices; and
4. Modernization of Current Barriers and Practices.

This approach ensures that all issues/initiatives are considered deterministically and in terms of their contribution to overall risk reduction associated with plant operation.

The matrix shown in Table D-6 provides a numerical value to be selected in Table D-5 that corresponds to major/minor, direct/indirect aspects of the impact utility assessment. The matrix comprises four types of improvements and four categories of contribution to establishing barriers for event-free operation.

For example, within this matrix, in relative terms, any improvement activity that results in achieving event free operation through physical improvements to the plant through new or augmented barriers and practices has the most "direct" and "major" impact on the cardinal objective of the Value Tree (safe plant operation and electricity production reliability). On the other hand, in relative terms, modernization of a current effective barrier or practice to an enabler could have an "indirect" and "minor" impact on the cardinal objective of the Value Tree.

Note that expressions of preferences are determined throughout by viewing initiatives in isolation without initially taking potential synergies or prerequisites into account. This is intentional to make prioritization tractable and to clearly identify the preference without subconsciously taking feasibility into account. Interdependence, synergy and feasibility are taken into account in the scoping and scheduling of corrective actions in the IIP.

The evaluation of each issue or initiative is conducted in two steps:

1. In terms of its best fit considering its contribution to defence-in-depth and improvement of barriers and practices assigning the value in the corresponding box.
2. If the contribution to any of the safety goals is judged to be significant (e.g., by an order of magnitude) then the value obtained in step one is automatically changed to 5.

The following should be considered in the use of Table D-5:

Values in boxes represent the Impact-Utility Score.

- **New** means there is no barrier or practice in place.
- **Augmentation** means current barrier is not complete or requires recovery or execution gaps in current practices or the modern codes and standards have complementary requirements that are not in place.
- **Improvement** means current barriers or practices are not kept fully effective.
- **Modernization** means current barriers or practices are effective but documentation and/or practices need to be updated to reflect current trends, state of the art approaches, terminology, etc., with no impact on operational performance.

Table D-5: Impact Evaluation Matrix

INSAG-10 Level of DiD	Row	Type of Issue/Corrective Action Requirement		Column: Nature of Improvement in Terms of Barriers and Practices			
				1	2	3	4
				New Barriers and Practices	Augmentation (Recovery) of the Current Barriers and Practices	Improvement (Effectiveness) of the Current Barriers and Practices	Modernization of Current Barriers and Practices
1	**1**	Maintain or Improve Design Basis (Includes Modifications or Replacements) & Implementation		5	5	4	2
1 & 2	**2**	Improve Operational or Safety Performance – (Plant Monitoring, Testing, Inspection & Maintenance, Configuration Management, Prevention of or Response to Events)		5	4	3	2
3 & 4	**3**	Reduce Uncertainty in engineered safety features and accident procedures– (e.g. Engineering reviews, studies, analysis, FFS, LCM, AMP, SOE)		4	3	2	1
1-5	**4**	Improve Managed System and Organizational Effectiveness (Process, program, procedure)	a. Field Impact (e.g., operating, outage, maintenance, field procedures)	4	3	2	1
1-5			b. Managed system and support processes Impact (e.g., general training, OPEX)	3	2	2	1

D.7. Quantifying the Utility of Issue Resolution

The time and impact attributes discussed in Section D.6 above express the preferences of the decision maker, but not uncertainty. To quantify the utility of resolving an issue, it is necessary to combine the time and impact attributes to obtain a numerical value that represents the utility of the two parameters each rated on their 1 to 5 scale. This is achieved through the use of utility functions such as the following:

$U(x) = 1 - e^{-x/R}$

where:

- U is the utility of attribute x, with $0 < U < 1$
- x is the score of the attribute on some scale (in this case, the 1 to 5 scale)
- R is an adjustable parameter that expresses preference; in this case, by adjusting R the utility is more strongly weighted towards solutions with high Impact or Time, versus simply the linear relationship provided by a 1 to 5 scale

Since the decision model expresses preference in terms of two attributes (Time and Impact) a two parameter utility function is formulated. Given utility independence, the following function is used:

$U(i,t) = k_i\, U_I(i) + k_t\, U_T(t) + (1-k_i-k_t)\, U_I(i)U_T(t)$

where:

- U(i,t) is the utility of resolving the Issue, taking into account the Time and Impact scores each assessed on a 1 to 5 scale
- k_i is the contribution of the Impact attribute
- $U_I = 1 - e^{-i/R_I}$
- k_t is the contribution of the Time attribute
- $U_T = 1 - e^{-t/R_T}$

To use the two–parameter utility function, the IIPPT evaluated its preferences for Impact (R_I) and Time (R_T), as well as $k_i = U(5,1)$ and $k_t = U(1,5)$. This evaluation resulted in the following values:

- $k_i = 0.3$
- $R_I = -5.0$
- $k_t = 0.1$
- $R_T = -2.0$

The result is the utility matrix given in Table D-6. To use the matrix, the Time and Impact values are determined on a 1 to 5 scale, and the resulting utility (score) associated with resolving the Issue is given by the number in the cell associated with the Time and Impact scores.

Table D-6: Utility Matrix

		Time				
		1	**2**	**3**	**4**	**5**
Impact	**1**	0.00	0.01	0.03	0.05	0.10
	2	0.05	0.08	0.11	0.17	0.26
	3	0.12	0.15	0.21	0.31	0.46
	4	0.20	0.25	0.34	0.48	0.70
	5	0.30	0.37	0.49	0.68	1.00

The table indicates that the IIPPT strongly prefers solutions that:

- Have greater impact on the objective in shorter time, i.e., (4,4) is 6 times more preferable than (2,2)
- Have larger impact versus one that can be done quickly but with little impact, i.e., (1,5) is three times more preferable than (5,1)

D.8. Using the Decision Model

D.8.1. Application of the Decision Model to the Global Assessment Methodology

Since the Value Tree provides an integrated, coherent set of values for the assessment of the Global Issues and the IIP, it is used to guide the following processes:

1. Initial Ranking of GIs and GIOs against the second tier of the Value Tree; and
2. Preliminary and Final Ranking of CAs against the third tier of the Value Tree.

D.9. Pairwise Comparison of the Value Tree Elements

D.9.1. Description of the Process and Preliminary Results

The weights will be assigned using the pairwise comparison technique that is used in the Analytic Hierarchy Process (AHP). This technique forces the decision maker to choose between two objectives at a time by ranking them in terms of their relative contribution to the higher level objective.

The pairwise comparisons are performed with the knowledge and understanding that Bruce B is to be operated for an additional 30 or more years, and that it must be operated safely and with high reliability to support electricity production per Bruce Power's business goals. The rankings used to distinguish between pairs of objectives are presented in Table D-7.

Table D-7: Ranking Definitions

Rank	Importance Descriptor
1	Equally
2	Equally – Moderately
3	Moderately
4	Moderately – Strongly
5	Strongly
6	Strongly – Very Strongly
7	Very Strongly
8	Very Strongly – Extremely
9	Extremely

For each pairwise comparison, a score is provided, based on these ranking definitions, followed by the justification for the score. As indicated in Section D.2, the IIPPT develops the pairwise comparisons, which are then used directly to determine the branch weights, per Section D.5.