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Bruce Power Acceptance Form

Bruce Power

CANDESCO

BRUCE A INTEGRATED SAFETY REVIEW BASIS DOCUMENT

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REV. 00

☒ Accepted
☐ Rejected
☐ Accepted As Noted – Revision Required
☐ Accepted As Noted – No Revision Required

FOR USE AT BRUCE POWER

ACCEPTED: **Frank Saunders** (Please Print Name) (Signature)

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DATE: **10OCT2014**

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Title: Bruce A Integrated Safety Review Basis Document

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A Report Submitted to Bruce Power

October 10, 2014

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Acronyms and Glossary

CAP	Corrective Action Plan
CARD	Corrective Action Requirements Definition
CDF	Core Damage Frequency
CED	Code Effective Date
CNSC	Canadian Nuclear Safety Commission
CSA	Canadian Standards Association
CSI	CANDU Safety Issue
EA	Environmental Assessment
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
LTEP	Long Term Energy Plan
MCR	Major Component Replacement
NPP	Nuclear Power Plant
OPEX	Operating Experience
PAISR	Periodic Application of Integrated Safety Review
PCA	Potential Corrective Actions
PIO	Potential Improvement Opportunity
PRA	Probabilistic Risk Assessment
PROL	Power Reactor Operating Licence
PSA	Probabilistic Safety Assessment (synonymous with PRA)
PSR	Periodic Safety Review
RD/GD	Regulatory Document/Guidance Document
RIDM	Risk-Informed Decision Making

SBR	Safety Basis Report
SCR	Station Condition Record
SFR	Safety Factor Report
SIP	Safety Improvement Plan
SRS	Systematic Review of Safety (synonymous with ISR)
SSCs	Structures, Systems and Components
TOE	Technical Operability Evaluation
WBS	Work Breakdown Structure

Corrective Action: A corrective action that used to be a PCA but that remains after ranking, prioritization and risk informed decision making activities have eliminated some other PCAs.

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 Corrective Action (PCA): A corrective action that would, if taken, partially or completely address a GIO. Each GIO may have a number of PCAs that will collectively address the complete improvement opportunity.

Potential Improvement Opportunity (PIO): Animprovement initiative that originates from the ISR process or that may already be planned or in progress having originated elsewhere. Other origins of PIOs include: the 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 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 ISR Basis Document for the Bruce A Integrated Safety Review (ISR) and sets out the scope and methodology for the ISR to be conducted by Bruce Power in 2014/15. The ISR Basis Document will be submitted to the Canadian Nuclear Safety Commission (CNSC).

This basis document will govern the conduct of the ISR 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 ISR. The basis document identifies:

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;
4. the operating strategy of the facility;
5. the methodology for the performance of the ISR, including the period for which the ISR is valid;
6. the applicable national and international standards, codes and practices;
7. the methodology for the global assessment;
8. the structure of the ISR documentation; and
9. ISR governance.

It also describes the process for categorizing, prioritizing, resolving, and tracking findings.

1.2. Background

Bruce Power, as an essential part of its operating strategy and as part of the Long Term Energy Plan (LTEP) (http://www.energy.gov.on.ca/en/ltep/), is planning to continue Unit 3 and Unit 4 operation well into the future. This will necessitate replacing major components such as pressure tubes, feeders and steam generators such that the units are maintained in a fit for service state. A Major Component Replacement (MCR) approach will be used to replace components as required over the life of the units. However, due to the unusually long outage and de-fuelled state during pressure tube replacement there is an opportunity to conduct other work and some component replacements that could not be done reasonably in a maintenance outage will be scheduled concurrently. Conduct of this ISR will support definition and timing of practicable opportunities for enhancing safety of Units 3 and 4. This review will also include the ongoing operation of Units 1 and 2, which have already been refurbished.

To achieve this Bruce Power is conducting a station-wide review of safety for Units 0A and 1-4 to be termed an ISR. This review will supplement the interim Periodic Safety Review (PSR) that was conducted for the ongoing operation of the Bruce A and B units until 2019 [1] since all 15 Safety Factor Reports (SFRs) (14 from SSG-25, plus radiation protection) will be prepared and the outlook will be further into the future. The Bruce A ISR will meet or exceed the international guidelines given in International Atomic Energy Agency (IAEA) Guide SSG-25, Periodic Safety Review for Nuclear Power Plants [2]. This ISR is being conducted as part of ongoing operation for Units 1 and 2 and to support MCR and asset management in Units 3 and 4, so it also envelops the guidelines in the current version of CNSC Regulatory Document RD-360, Life Extension for Nuclear Power Plants, issued February 2008, with the exception of those related to the Environmental Assessment (EA), which has already been completed for Bruce A [3].

It is recognized that RD-360 is in the process of being superseded by CNSC REGDOC-2.3.3 [5], Integrated Safety Reviews. However at the time this ISR Basis was prepared REGDOC-2.3.3 was still in draft form. Nevertheless, given that the draft states that the REGDOC is consistent with SSG-25, it is expected that overall the ISR will be consistent with REGDOC-2.3.3 once the latter is issued. No significant changes to the ISR are anticipated as a result of the formal issuance of REGDOC-2.3.3.

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

1. ISR Basis Document (this document);
2. Reports on the review of each safety factor (SFRs);
3. Global Assessment Report (GAR); and
4. The Final Report and Integrated Implementation Plan (IIP).

The deliverables from this ISR will also provide a sound basis for similar reviews in support of future PSRs at Bruce A and Bruce B[1].

1.3. Scope and Structure of Document

The rest of this basis is structured as follows:

- **Section 2: ISR Scope**, describes the objectives of the ISR and the overall scope of the ISR effort. It describes the current licensing basis and identifies the Safety Factors to be reviewed with reference to SSG-25 and refers the reader to Appendix A for a detailed exposition of the review tasks of SSG-25. 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 ISR.
- **Section 3: ISR Methodology** starts out by describing the systematic approach to the ISR that will be followed before delving into the details of the three major phases of the

[1] When completed, the full PSR is expected to be referred to as a Periodic Application of Integrated Safety Review (PAISR).

ISR endeavour: 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 way in which findings of the review tasks will be categorised is also described. 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 and MCR in Bruce A. 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 subjected to Risk Informed Decision Making (RIDM), where necessary, and plans formulated to implement the selected improvements to arrive at an IIP.

- **Section 4: Recording the Output of the ISR** notes that a database will be used to support and record the results of the execution of the ISR before describing the content of the main outputs of the ISR: SFRs, GAR, and Final Report and IIP.
- **Section 5: Managing the Execution of the ISR** describes Bruce Power's organization for the ISR, the overall time-frame with major milestones as well as quality assurance provisions.

2. Scope of the ISR

2.1. ISR Objectives

According to IAEA SSG-25, Periodic Safety Review of Nuclear Power Plants [2], a PSR/ISR can be used for a number of purposes, i.e., as a systematic safety assessment carried out at regular intervals, in support of the decision making process for licence renewal, or in support of the decision-making process for long term operation.

The safety of Bruce A 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 current licensing basis requirements 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 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.

Bruce Power's position is that the current and near-term safe operation of Bruce A is assured by these existing processes. Therefore, the ISR is not a vehicle to assess the current safety of Bruce A. Rather, the ISR is a vehicle to assess the medium to long-term prospects for safe operation of Bruce A 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 (Section 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 the ISR as complementary to the existing processes, as it provides a vehicle to look forward on approximately a 30 year timeframe, which is longer than the timeframes typically associated with licence renewal and the other processes that focus on near-term safety.

The overall objective of the Bruce A ISR is to obtain a practicable set of work to be conducted during the MCR in Units 3 and 4, and during asset management activities to support ongoing operation of all four units, that will enhance safety to support long term operation. The look-ahead period will be longer than the interim PSR that was done for Units 1-8 [6]. It will cover a 10-year period, as there is an expectation that a PSR will be performed on approximately a 10-year cycle, given that Units 3 and 4 could be operated well into the future. Nuclear Safety is a primary consideration for Bruce Power and the management system must support the enhancement and improvement of safety culture and the achievement of high levels of safety as well as business performance.

The specific objectives of the ISR 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 A (by definition, the licensing basis is always valid, so this objective from SSG-25 and RD-360 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 Structures, Systems, and Components (SSC) and programs that are in place to ensure plant safety for long-term operation; and
4. Determine the practicable improvements to be implemented to resolve any findings identified in the review and timelines for their implementation.

2.2. ISR Scope

The scope of the ISR is to conduct a review of Bruce A that meets the elements given in IAEA Guide SSG-25 [2] and CNSC Regulatory Document RD-360 [7] issued in February 2008, with the exception of the RD-360 guidelines related to the EA.

The work on this ISR shall build on the extensive body of previous work as follows:

- The Systematic Review of Safety conducted for the Units 1 and 2 refurbishments circa 2006 and follow-up assessments including the resolution of Category 1, 2, and 3 items;

- The ISR Basis Report, 14 SFRs, IIP, and GAR, conducted circa 2008 for the proposed Unit 3 and 4 refurbishments [8][9][10][11];
- The Safety Basis Report (SBR) prepared by Bruce Power and issued to the CNSC via letters NK21-CORR-00531-11005 and NK29-CORR-00531-11397, dated December 30, 2013 [6]; and
- Other ongoing CNSC issues, such as Action Items, Fukushima Action Items (FAI), CANDU Safety Issues, and licence submissions.

An EA was performed as stipulated by the Canadian Environmental Assessment Act, based on the refurbishment of all four Bruce A units. The EA was submitted and found acceptable by the CNSC as documented in the Record of Proceedings July 5, 2006 [12]. Therefore, an EA is not required in the scope of this ISR.

The scope of the ISR can therefore be summarized as the following:

- Performing assessments of the Bruce A plant and Bruce Power governance against a set of modern codes and standards;
- Using the results of the assessments, evaluate the Bruce A plant and Bruce Power governance against 14 Safety Factors prescribed in SSG-25, plus an additional Safety Factor to encompass Radiation Protection;
- 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 reasonable and practicable plans for the improvement opportunities while taking into account previously developed station improvement plans to arrive at a single coherent IIP for Bruce A.

2.3. Safety Factors to be Reviewed

In SSG-25 [2], the IAEA identifies five Subject Areas which 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.

The safety reviews for Bruce A will be compiled in the 15 safety factor reports shown in Table 1. The intent is to follow the format of SSG-25 [2], augmented as required to include the additional topics recommended by RD-360 [7].

In carrying out the ISR, each Safety Factor is comprised of review tasks as recommended in Reference [2], and suggested by Reference [3]. They are listed in Appendix A. The concept underlying the ISR 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 ISR is provided in Appendix A. Note that SSG-25 [2] 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 A design and operation.

In particular, the scope of the ISR shall address Systems Important to Safety as defined by Bruce Power in compliance with CNSC document RD/GD-98, "Reliability Programs for Nuclear Power Plants" [13] and systems defining the Safe Operating Envelope.

Table 1: Scope of the ISR

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 measures implemented in the plant design

The ISR shall also include 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 (CSI); and
- Bruce Power Regulatory Commitments.

2.4. Statement of Current Licensing Basis

Both SSG-25 and RD-360 require that the ISR Basis Document include a statement of the current plant licensing basis. The term “licensing basis” is defined in CNSC document INFO-0795 [14] 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) [15], 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 [16] (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 2008 [17] in support of the current operating licence, from supplementary submissions in 2008 [18] and 2009 [19], and from additional documents

identified in the LCH as being part of the licensing basis [20][21]. In addition to the information supporting the current licence, Table B-1 also lists the current Bruce Power governance documents included in the 2013 licence renewal application.

2.5. Statement of Modern Codes, Standards and Practices

Bruce Power has in the past conducted assessments of the plants and processes against a large number of codes and standards [8] [22] [23]. This section elaborates on the scope of the codes and standards applicable to the Bruce A ISR, and which modern codes and standards will be assessed.

One of the key elements of the ISR 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 point of departure for identifying codes and standards to be assessed for the ISR is the review tasks identified in Section 5 of SSG-25 [2]. These were adjusted as necessary for the unique circumstances of Bruce A and are included as Appendix A. In cases where additional guidance was deemed necessary the review tasks were augmented with information culled from sections on methodology from Section 5 of SSG-25 [2].

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 applicable Canadian regulatory documents (i.e., CNSC Standards, Requirements and Guides) that are not already included in the current licensing basis. The next step was to identify applicable non-regulatory 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 ISR 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 August 31, 2014, are considered for review in this ISR.

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 in the operating licence and to transition into compliance with codes and standards that are expected to appear in the next licence. 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 ISR.

Codes in Table C-1 which are referred to in the Bruce A Operating Licence-PROL 15.00/2015 have already, or are being, transitioned into Bruce Power governance, so no assessments of these codes are required. If a code or standard has been updated since the current licence, then the Safety Factor authors should refer to the new Application for the Renewal of the Bruce A Power Reactor Operating Licence (PROL) [27] to identify if a transition plan is available.

Some codes in Table C-1 above may be gradually transitioned into Bruce Power governance, while the remainder are not likely to be transitioned into Bruce Power governance, but serve as a basis for assessing the review tasks in Appendix A. The codes and standards in Table C-1 have either been previously assessed, or will be assessed in this ISR. Gap assessments of the codes and standards in Table C-1 that have already been completed will be used in the ISR and their validity confirmed. An assessment will be performed where no gap assessment presently exists.

3. ISR Methodology

3.1. Points of Departure

According to SSG-25 [2] the methodology for an ISR should include making use of safety reviews that have already been performed for other reasons. In the case of the Bruce A ISR this includes reviews that were conducted for the following purposes:

- Return to service of Bruce A Units 3 and 4 [1] (circa 2001);
- Life extension of Units 1 and 2 (circa 2006);
- Proposed refurbishments of Units 3 and 4 and Bruce B (circa 2008); and
- SBR [6] and PSR for Units 1-8 (2013).

These reviews covered many, if not all, of the same Safety Factors that need to be reviewed. The following serves as the basis for the Bruce A ISR methodology:

1. There is a common licensing basis for Bruce A Units 1 and 2 and Bruce A Units 3 and 4. That is, the work done for the Bruce A Units 1 and 2 life extension project shall be reviewed for application to the Bruce A Units 3 and 4 MCR project dependent on actual plant condition. The Bruce A Units 1 and 2 Systematic Review of Safety (SRS) Basis document [23] describes what was done in the Bruce A Units 1 and 2 SRS review;
2. The same ISR basis will apply to Bruce A Units 1 and 2 and Bruce A Units 3 and 4. The modern codes and standards listed in Appendix C are the basis for this ISR;
3. The IIPs from the earlier ISRs and PSRs will serve as a reference for the resolution of gaps and planned safety improvements. The Bruce A ISR will use these as input for the Safety Factor reviews;
4. Each Safety Factor review will address those elements listed in Appendix A; and
5. 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 scanned for existing deficiencies and planned corrective actions.

3.2. Systematic Approach to the ISR

The approach to the execution of the ISR is systematic both in the sense that it is comprehensive and that it follows 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 (GIO). Next corrective actions are identified for each GIO, before risk-informed decision-making 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.

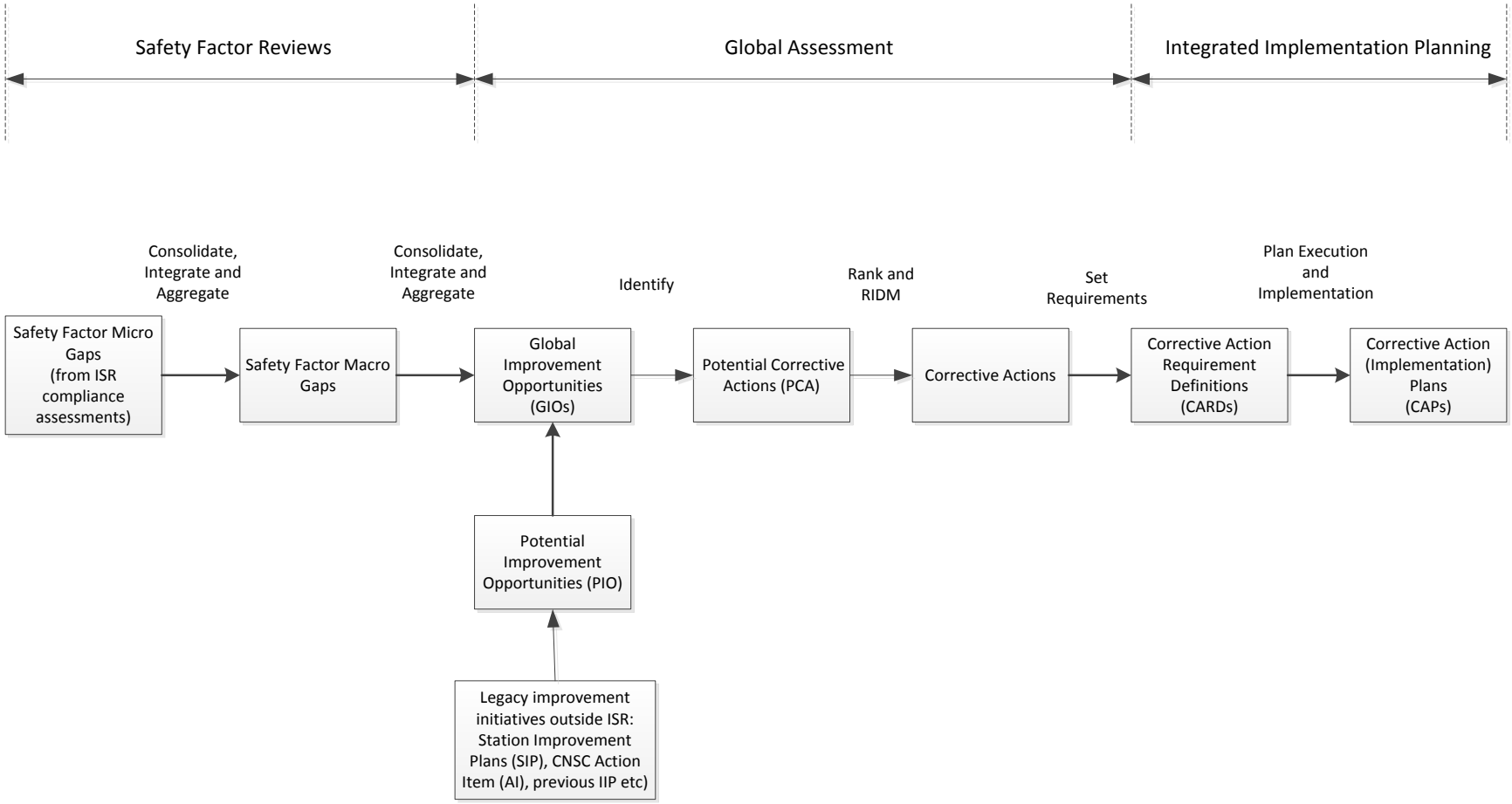

Figure 1: Integrated Safety Review Process

3.3. Conducting Safety Factor Reviews

This generic review process of safety factors described in SSG-25 has been tailored to proceed along the following steps for the Bruce A ISR:

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 alignment with the provisions of the review tasks;
6. Identify 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 ISR Basis document, the *Objective*, *Scope and Tasks*, and *Methodology* guidance provided in Section 5 of SSG-25 [2] has been adjusted as necessary for Bruce A. The results of this are 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 focus is precise and specific.

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 should 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 arguments for why the subject document will, or will not be, assessed as well as arguments for the type of assessment (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 should 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 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. 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, such as pressure-boundary components, and that provide detailed requirements for their design, construction and maintenance. In

this case assessments are performed to identify gaps that could result in future design changes depending on the outcome of the RIDM process.

3.3.4. Perform Gap Assessments against Codes and Standards

This step provides for the actual assessment of the Bruce programs and the Bruce A plant against the identified codes and standards. In general this involves determining from available design or programmatic documentation whether the plant's design or programs meet the provisions of the specific clause of the standard or of some other criterion 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 should be in the form of the assessor's arguments conveying reasons 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 – compliance has been demonstrated with the applicable clause;
- Indirect Compliance – Compliance has been demonstrated with the intent of the applicable clause;
- Acceptable Deviation – Compliance with the applicable clause cannot be demonstrated; however, a technical assessment has determined that the deviation is acceptable. For this case a detailed discussion and explanation shall be included in the ISR documentation;
- Gap – system design and/or operational improvements may be necessary. Gaps are categorized as follows to facilitate later assessment and consolidation:
    - Programmatic: Resolution of the gap may require changes to process, methods and operating procedures;
    - Engineering: Resolution of gap may require modification to the facility or changes to design documentation;
    - Analytical: Resolution of gap may require nuclear safety analysis (deterministic or probabilistic);
    - Effectiveness: There is evidence that an implementation gap exists, based on a review of audits/assessments, etc.;
- Guidance: A potential programmatic, engineering, analystical or effectiveness gap found against non-mandatory guidance;Relevant but not Assessed – the particular clause provides requirements that are less strenuous than clauses of another standard that has already been assessed; and
- Not Relevant – 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.
- Not Applicable – The text is not a clause that provides requirements or guidance. Also used if the clause does not apply to the specific facility.

Notwithstanding the compliance categories above if, in the course of these assessments, it is found that the Bruce A 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 or Technical Operability Evaluation (TOE) processes. 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 non-mandatory guidance on how specific requirements may be met. The treatment of such non-mandatory guidance for the Bruce A ISR is based on consideration of the following:

- It is recognized that a reason for performing an ISR 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 non-mandatory 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 cost.
- 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.

Given the considerations outlined above, assessment against non-mandatory guidance in codes and standards will adhere to the following principles:

- 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;
- Non-mandatory guidance that does not directly support the execution of safety factor review tasks will, in general, not be assessed. However this principle should be balanced at the discretion of the assessor with the need to identify reasonable safety improvements which may not be review task related;
- 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 was found against guidance or a requirement.
- In cases where guidance comes in the form a whole section of a document, large portion of text or appendix a high-level assessment may be performed at the discretion of the assessor.

In the majority of cases the Bruce A ISR will not be the first time that the Bruce A 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 ISR
1	Compliance against the clauses of the standard was previously assessed and the standard has not been replaced by a new version before the code effective date (CED) of August 31, 2014.	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 CED of August 31, 2014 and the code will not be referenced in the new PROL.	Confirm that the previous assessment is still valid for unchanged clauses by updating references to programmatic documents. Assess compliance with updated clauses.
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 CED of August 31, 2014 and the code will not be	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.

Scenario	Nature of Change in Standard	Assessment Approach for ISR
	referenced in the new Power Reactor Operating Licence.	
4	The standard is new having been issued before the CED of August 31, 2014 and a clause-by-clause assessment was not previously performed and the code will not be referenced in the new PROL.	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 2.

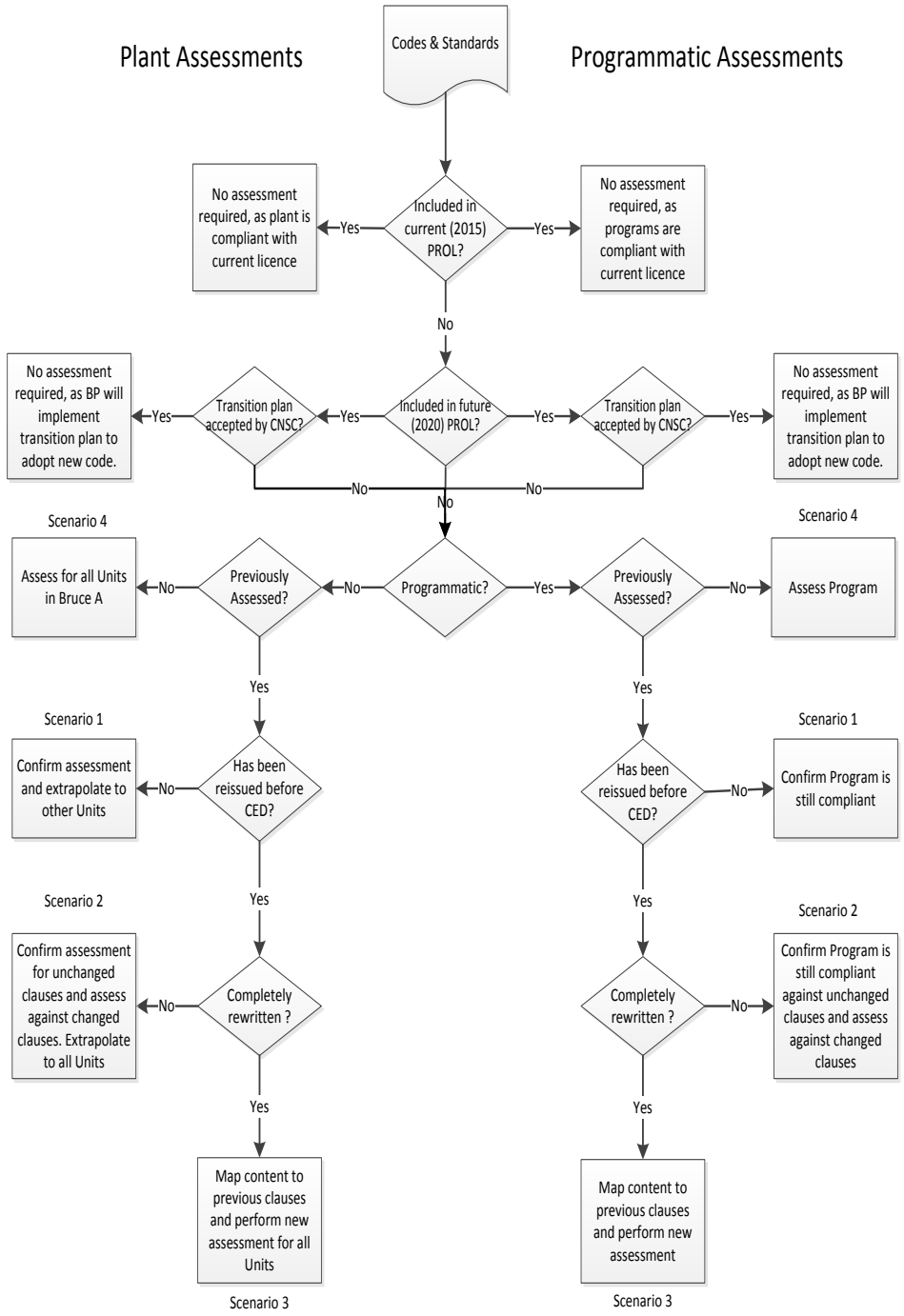

Figure 2: Assessment Scenarios for Clause-by-Clause Assessments

3.3.4.2. Plant Clause-by-Clause Assessments

Figure 2 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 ISR
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 CED of August 31, 2014.	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 CED of August 31, 2014 and the code will not be referenced in the new PROL.	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 CED of August 31, 2014 and the code will not be referenced in the new PROL.	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.

Scenario	Nature of Change in Standard	Assessment Approach for ISR
4	The standard is new having been issued before the CED of August 31, 2014 and a clause-by-clause assessment has not been performed previously and the code will not be referenced in the new PROL.	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 ISR. The results of this assessment are documented in Section 5 (Overview of Bruce A Station Programs and Processes) of each SFR (see Section 4.2 herein).
- An **Implementation and Effectiveness Assessment** in which all independent audits, self-assessments and regulatory evaluations and reviews pertaining to the particular program and period under consideration 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 A ISR provided 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 CED;
  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 since the Bruce 3 and 4 ISR in 2008 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 can be included in the design, construction, and maintenance processes for new or replacement components, should the RIDM process support this outcome. Existing equipment is grandfathered , the assumption being that itmeets 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 [24] 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 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 clear evidence that the Bruce A plant or programs by far exceed mere 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 known as safety factor macro-gaps which are listed in Section 8 (Summary and Conclusions) of each SFR (see Section 4.2).

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 generic global assessment process described in SSG-25 and Draft REGDOC-2.3.3 has been tailored to proceed along the following steps for the Bruce A ISR:

1. Consolidation of Safety Factor Findings;
2. Developing an Assessment Framework;
3. Prioritization and Ranking of GIOs;
4. Identification of Potential Corrective Actions (PCA);
5. Prioritization and Ranking of PCAs;
6. Perform RIDM;
7. Global Assessment; and
8. Documentation of the results in the GAR.

Each of these steps is discussed in more detail below.

3.4.1. Consolidation of Safety Factor Findings

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

- Summarize the safety factor report findings and identified gaps and strengths;
- Address any overlaps, omissions, and interface issues of the findings from the SFRs; and
- Consolidate macro-gaps into global issues where appropriate.

The findings from each safety factor review, be they strengths or macro-gaps, are based on the fairly narrow perspective of the safety factor. This step of global assessment provides for the consolidation of these findings into 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 macro-gaps. In addition other legacy issues that have been identified through other assessments or initiatives outside the ISR will be integrated with the safety factor micro-gaps to form a comprehensive list of Global Issues that need to be addressed. The consolidation of safety factor findings therefore comprises the following:

- **SFR Gap Consistency Checking** involves the following:
  - Identifying all codes and standards that have been assessed by more than one Safety Factor;
  - For each code or standard that was assessed by multiple Safety Factors, identifying which clauses have been assessed by which Safety Factors;
  - Screening of the assessment categories of clauses with multiple assessments in the various SFRs for duplication, completeness, and consistency;
  - Identifying any inconsistencies as a new issue for inclusion in the database under the category of *Consistency Gaps*; and

- Identifying, where possible, measures to be taken to rectify the inconsistency, or alternatively, indicating that further investigation is needed.

- **SFR Gap Consolidation** involves the following:
  - Starting with the first SFR Gap (i.e., macro-gap) identified for Safety Factor 1, scrutinizing the remainder of the gaps for coverage of the same topic by using the search features of the database;
  - If duplication is found:
    - Linking, in the database, the affected SFR Gaps to a single PIO and associating with the new PIO the micro-gaps associated with the SFR Gaps being consolidated; and
    - Providing a clear PIO title and description that is as specific as possible to ensure it covers the associated micro-gaps across all affected Safety Factors.
  - If no duplication is found:
    - Creating a PIO that corresponds to the SFR Gap; and
    - Reviewing and modifying the PIO title and description as necessary to ensure it is as specific as possible and covers the associated micro-gaps across all affected Safety Factors.
  - Identifying any micro-gaps that are orphaned as a result of the steps above for inclusion in the Master Issues List in the database under the category of *Consistency Gaps.*
- **Collect and Integrate all non-SFR Gaps and Issues** involves the following:
  - Identifying the sources of potential issues that may have to be included in the Master Issues List in addition to the SFR micro-gaps. The Master Issues List will include, but not be limited to, issues originating from the following past assessments and ongoing activities appropriately cross-referenced to their original sources:
    - CNSC Action Items (AI);
    - Fukushima Action Items(FAIs);
    - CANDU Safety Issues (CSIs);
    - Licence submissions, etc.;
    - Orphaned issues resulting from consolidation activities themselves.
    - Station Improvement Plans (SIPs); and
    - Previous Integrated Implementation Plans (IIPs).
  - Where needed, obtaining access to the sources of these issues;
  - Collecting all these issues together in the database while retaining cross-references to their source of origin; and
  - Consolidating these additional issues by assigning those that belong together under a single PIO created for that purpose.
- **Consolidate PIOs from SFRs and Other Sources into GIOs:** The purpose of this step is to consolidate PIO initiatives from the SFRs and other sources such as the SIP that

are related to each other or that overlap in intent and/or scope. Collections of PIOs and other initiatives will be grouped together under entities known as GIOs. In order to remove the potential for duplication of future work, a check will be done for any additional project, CNSC AIs, etc., that had not already been identified as PIOs, but that are relevant in support of the development of CAPs. Those PIOs identified as duplicates are closed and the supporting rationale will be provided. The outcome of this step will be a compilation of the needs and activities in progress which will be used as the basis for the corrective action planning. Those PIOs that have been closed at the end of this step will be tabulated in the IIP Report.

3.4.2. Developing an Assessment Framework

The objective of developing an assessment framework 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.

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

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

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

More specifically, a process is needed to decide on the importance ranking and prioritization of the issues identified through the ISR 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 compliance with 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 at a 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 will be imbedded in the database. It is likely that the value tree will have two tiers below the cardinal objective for a total of three tiers.

Given the difference in objectives between Bruce 1 and 2 (continued safe operation) and Bruce 3 and 4 (life extension through MCR and asset management, together with continued safe operation), separate assessment frameworks may have to be devised for Bruce 1 and 2 and Bruce 3 and 4.

3.4.3. Prioritization, and Ranking of Global Improvement Opportunities

The purpose of this step is to arrive at a list of GIOs ranked in order of priority to resolve through corrective action 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 not the feasibility of the associated corrective actions subject to constraints of cost and time or intangible considerations. The latter is part of development of the IIP. The ranking and prioritization step entails the following:

- Use the GAF described in Section 3.4.2, as implemented in the database, to assign each GIO to a second tier objective in the value tree. In so doing, the Global Issue assumes the same priority as the objective as expressed in the weight of the objective;
- Taking into consideration the nature of potential corrective actions for the GIO use the GAF to evaluate the impact and time-to-take-effect of resolving the GIO. In so doing, a two parameter utility score is assigned to the GIO;
- Calculate a ranking number for the GIO by multiplying the assigned weight and score; and
- Arrange the GIO based on ranking number from highest to lowest to arrive at a ranked list.

3.4.4. Identification of Potential Corrective Actions

This step provides for the identification and high level definition of PCAs to address each of the GIOs. This may involve grouping GIOs together since a single PCA may address more than one Global Issue or it may be necessary to devise more than one PCA for the same Global Issue. The development of the PCAs will adhere to the following principles:

- An integrated approach will be taken to remove scope overlaps and optimize available time and resources – corrective actions identified either through the SFRs or the other sources such as the SIP should be integrated and consolidated;

- Deterministic and probabilistic safety assessment insights (e.g., where applicable, contribution to Core Damage Frequency (CDF) or safety goals or reduction in public dose, etc.) will be utilized to the extent practicable in establishing risk importance, prioritization, and ranking of improvement opportunities that will be subject to a RIDM process;
- Contribution of corrective actions to defence-in-depth and the fundamental safety functions will be taken into consideration;
- Corrective actions to be taken for the implementation of each improvement opportunity will be evaluated in terms of its contribution to actual benefit to safety taking into consideration how soon it will be effective once implemented;
- Alternative means of achieving the safety benefit will be 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;
- Ranked and prioritized corrective actions will be further integrated to optimize available resources and time and to maximize the safety benefit; and
- Unit or station specific initiatives will be specified accordingly.

3.4.5. Prioritization and Ranking of Potential Corrective Actions

The prioritization and ranking of PCAs uses the GAF and follows the same process as that of GIOs, the only difference being that PCAs will be assessed against the third tier of the value tree. The ranking and prioritization step therefore entails the following:

- Use the GAF described in Section 3.4.2, as implemented in the database, to associate each PCA with the second tier objective in the value tree that corresponds to the branch associated with the highest ranked GIO it is intended to address;
- Assign the PCA to the appropriate third tier sub-objective that the PCA will support under the same second tier branch. In so doing, the PCA assumes the same priority as the sub-objective as expressed in the weight of the sub-objective;
- Taking into consideration the nature of the PCA 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 PCA;
- Calculate a ranking number for the PCA by multiplying the assigned weight and score; and
- Arrange the PCAs based on ranking number from highest to lowest to arrive at a ranked list.

3.4.6. Perform Risk Informed Decision Making

The need to perform a RIDM assessment will be established based on the scope, schedule and cost considerations defined in the implementation plan. For example, a RIDM assessment would be required in cases where:

- The schedule for completion extends beyond the proposed next licence period; or
- The implementation schedule overlaps with planned component/equipment replacement outages; or
- The associated costs are extensive such that implementation of higher ranked PCAs 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 [25]. The results of each RIDM assessment will be included in the GAR as an Appendix.

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

3.4.7. Global Assessment

The set of global strengths and global issues resulting from the consolidation of safety factor findings and the identified practicable Corrective Actions 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 A station through MCR and asset management for life extension. 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 corrective actions and safety improvements into account, together with their relative importance as expressed by their ranking numbers; and
- An assessment of the overall acceptability of operation of the Nuclear Power Plant (NPP) over the applicable period of the ISR and over the longer term outlook period.

3.4.8. 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.

3.5. Preparing the Final Report and Integrated Implementation Plan

The purpose of the IIP is:

1. To document planning for all of the corrective actions that will be implemented to address gaps identified in the ISR, based on their relative importance; and

2. To specify the schedule for implementing the resulting corrective actions and safety improvements.

The IIP will include the following:

1. The processes used for determining the detailed scope, including prioritization and scheduling of corrective actions and safety improvements;
2. The process and the associated methodology used to ensure that corrective actions and improvements that have the greatest impact on safety are completed in a timely manner;
3. Processes to be used for identification and management of project risks and controls;
4. Processes to be used to track the progress and completion of the corrective actions and safety improvements;
5. The process and the database used that demonstrates traceability providing appropriate references to the GAR; and
6. The basic principles for the change control process should it be necessary to update the planning in the IIP.

The IIP will be submitted to the CNSC.

The Corrective Actions identified during Global Assessment may be new or may involve previously identified or ongoing activities such as those included in the SIP. 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.

The development of the IIP entails the following steps:

1. Develop a High Level CARD for each corrective action;
2. Develop an Implementation Plan for each corrective action;
3. Optimize the IIP; and
4. Prepare Final Report and IIP.

Each of these steps is discussed in more detail below.

3.5.1. Develop a High Level Corrective Action Requirement Definition for Corrective Action

The preparation of CARDs entails the following:

- Establishing corrective action requirements to address all gaps (individual or consolidated) and their associated GIOs;
- Development of CARDs for practicable corrective actions and safety improvements by:
    - Describing the objective of the corrective action and clearly identifying the GIOs it is intended to address;

- Identifying the set of corrective action requirements to ensure that the corrective action addresses the GIOs as intended;
- Identifying outputs of the corrective action; and
- Identifying closure criteria that can be used to determine if requirements have been addressed adequately.

In this step PIO initiatives consolidated under GIOs from SFRs and other sources are integrated into a set of corrective actions to enable planning of their implementation. A high level CAP will be prepared that 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, Regulatory commitments, etc., will also be identified as part of this step.

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 developing the CAPs. Such documents can be used as the CAP when deemed appropriate.

The high level CAP will identify:

- Objective(s);
- An integrated set of corrective action(s) in the form of a Work Breakdown Structure (WBS) 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 units;
- A description for each corrective action;
- References to project plans or action tracking actions; and
- Any long lead aspects in the planning of corrective actions.

3.5.2. Develop an Implementation Plan for Each Corrective Action

A high level implementation plan which defines the work and schedule will be prepared for each corrective action. This high level Implementation Plan and Schedule should include:

- List of corrective actions based on the WBS from step 3.5.1;
- Where applicable, a sequence of corrective actions and the prerequisites of the work needed to be performed to complete the corrective action;
- The time period during which the corrective action will be performed;
- If available the following information may be included in the IIP but, alternatively may be included in a number of Project Execution Plans that support the IIP:

- Resource estimates (labor, materials, need for purchased services) for each activity;
- Duration and prerequisites for each activity; and
- Cost estimates for each activity.

In order to minimize potential duplication and the effort associated with preparing high level implementation plans, Action Tracking actions, 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.

3.5.3. 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 integrated review at both the CAP and associated corrective actions levels will be performed to remove potential duplication, identify opportunities for optimization of scope, resource needs and schedule.

3.5.4. Prepare Final Report and IIP

The results of the steps outlined above will be documented in a Final Report and IIP. The content of the Final Report and IIP is described in Section 4.4.

The final summary report shall be prepared 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 ISR are not achieved, and a list of areas where they are exceeded (that is, plant strengths);
- A summary of the outcomes from the global assessment; and
- An IIP of proposed safety improvements, including their safety significance and prioritization.

4. Recording the Output of the ISR

4.1. Database

The successful execution of an ISR 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 ISR to ensure that the approach and process are systematic. The database will serve as the single integrated data repository of all ISR activities, and will support the execution of the ISR 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 PCAs 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 PCAs; and
  - The database provides for numerous reports that facilitate the sorting of strengths and gaps to facilitate the global assessment activity.
- For Integrated Implementation Planning:
  - The database provides for definition of the CARDs associated with each corrective action; and
  - The database facilitates the development of the implementation plan for each corrective action.

4.2. Contents of the Safety Factor Reports

To facilitate the integration and reporting of the ISR 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 below.

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 A 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 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

Bruce A Reactor PROL.

3.3 Regulatory Documents

For example, CNSC RD-346.

3.4 CSA Standards

This includes the mandatory CSA standards cited in the Operating Licence 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 A Station Programs and Processes

This section lists the Bruce A programs, procedures and other station documents from 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

This section identifies areas where the review tasks have overlap with other Safety Factors and elements. It will include a description of the completeness of the overlapping reviews.

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

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 will include the following recommended by SSG-25:

- Significant ISR 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 ISR applicability period and for the 30-year outlook of this ISR.

4.4. Contents of the Final ISR Report and IIP

The Final Report and IIP will include the following material, consistent with SSG-25, Appendix II, Section II.6:

1. Summary of the outcomes of the SFRs;
2. Summary of the outcomes of the GAR, including:
   - Identification of negative findings arising from deviations between the present state of the plant and current safety standards and operational practices;
   - An evaluation of the safety significance of these negative findings; and
   - An overall judgment on the acceptability of continued plant operation.
3. The IIP, including CAPs for resolving negative findings by safety improvements or corrective actions, and their safety significance and priority;

   To ensure success the IIP will also specify:
   a. Organizational arrangements in place to execute the IIP;
   b. Governance applicable to the delivery of the IIP;
   c. Where necessary, scope, schedules and dependencies, for the earlier tasks that have an impact on critical path;
   d. A high level definition of resources and a resourcing plan;
   e. The mechanism for overall integration, peer or independent review and oversight; and
   f. 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.
4. An assessment of the safety of future plant operation over the period addressed in the ISR and for the 30-year outlook period of the ISR.

5. ISR Project Management

Bruce Power will prepare a project management plan to address the ISR project management. That document will include the project organization (see Figure 3).

The ISR project schedule is shown in Figure 4.

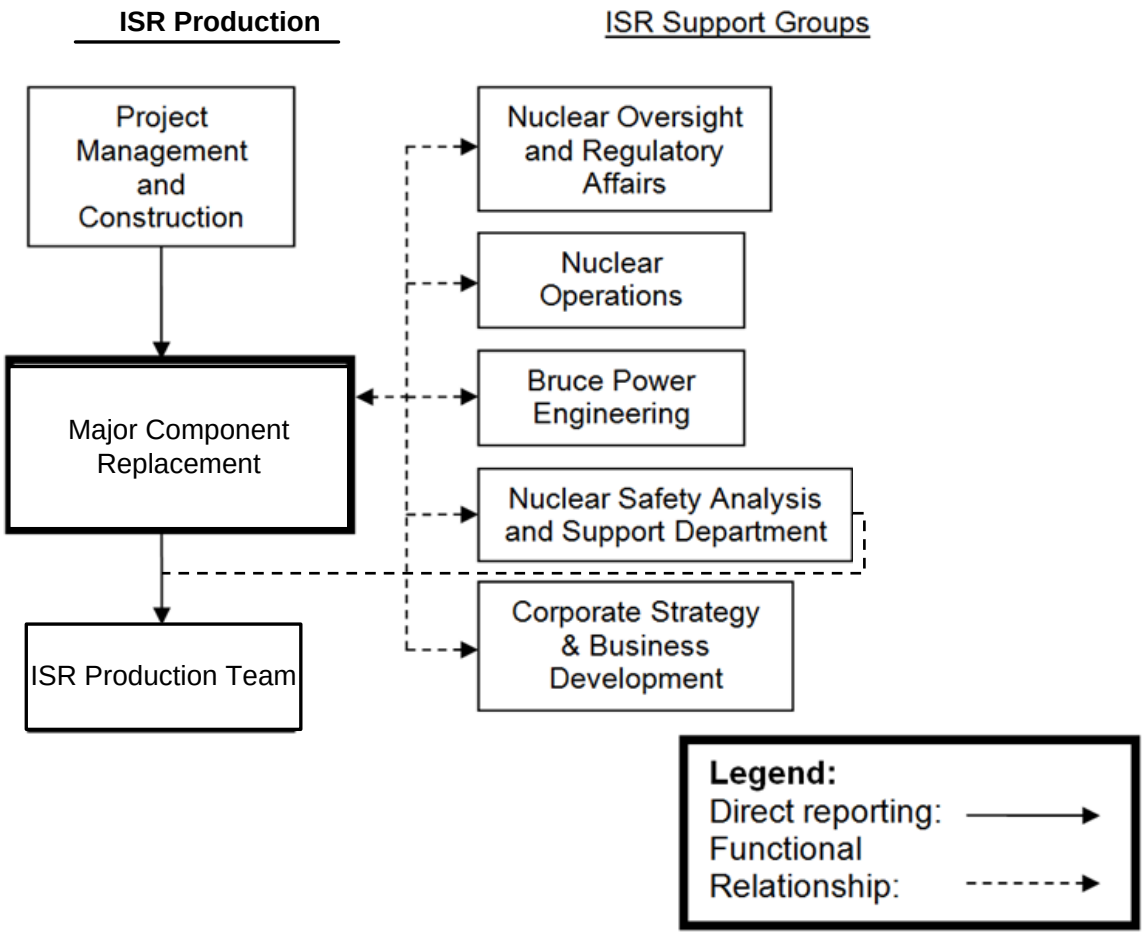

Figure 3: Bruce A ISR Project Organization

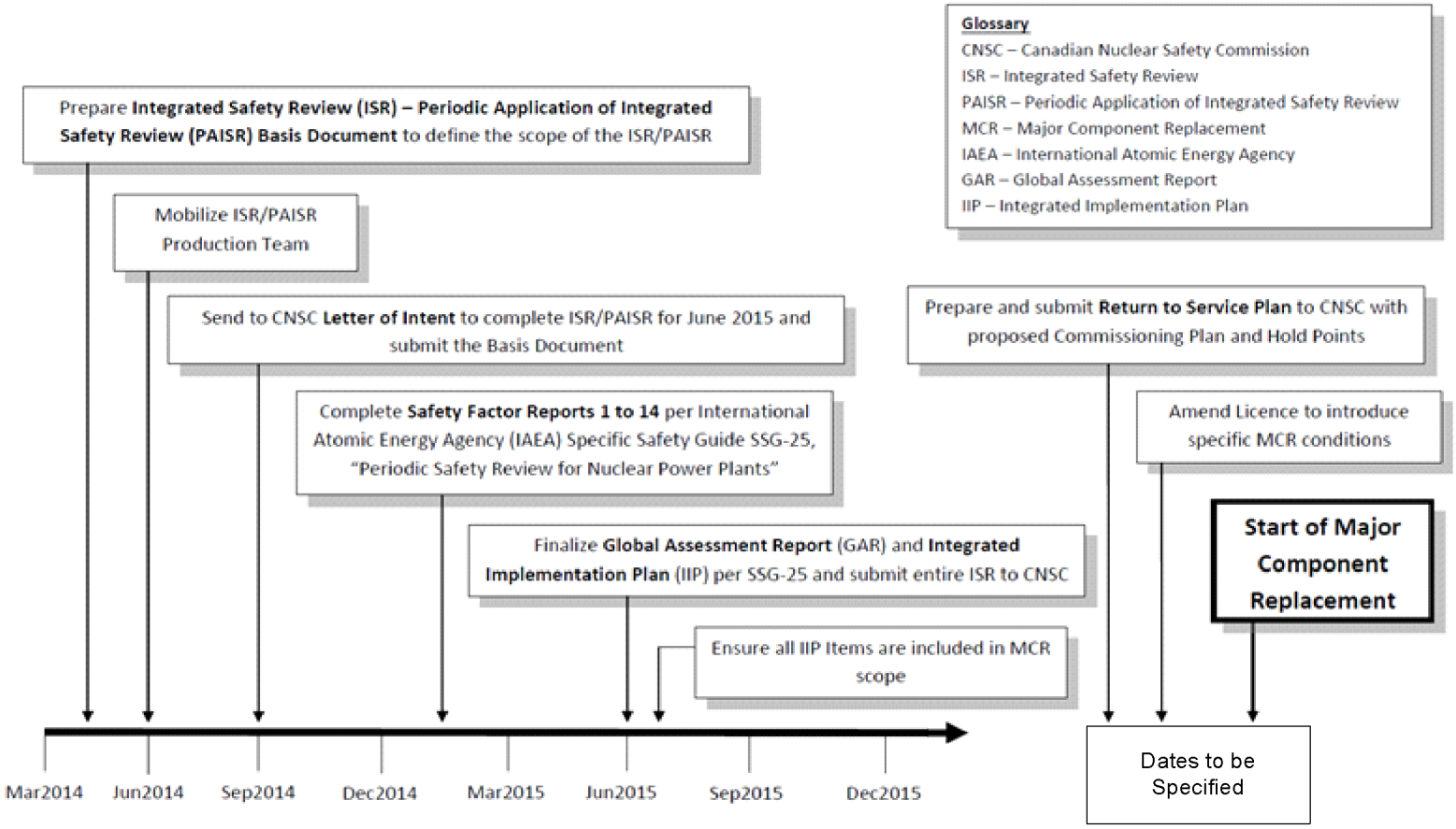

Figure 4: Bruce A ISR Project Schedule[2]

[2] When completed, the full PSR is expected to be referred to as a PAISR.

Further aspects of project management are discussed below.

5.1. ISR Project Quality Assurance

ISR work prepared by Bruce Power shall be prepared under the Bruce Power Management System Manual, BP-MSM-1 [26] and CSA N286-05 (Update No. 1), Management System requirements of NPPs.

Contractors' quality assurance program shall meet the requirements of ISO 9001.

5.2. Internal Project Communications

Project progress and issues will be communicated between the ISR Production Team and the ISR Project Management group through weekly progress meetings. In addition, the ISR Production Team will conduct separate weekly internal project meetings. Urgent issues will be communicated through direct contact between the ISR Production Team's project manager and the Bruce Power ISR 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 ISR 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 safety factor reports, the GAR, the Final ISR Report and the 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.

6. References

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

[2] IAEA Specific Safety Guide, “Periodic Safety Review for Nuclear Power Plants”, SSG-25, 2013.

[3] IAEA NS-G-1.13, "Radiation Protection Aspects of Design for Nuclear Power Plants", 2005.

[4] Bruce Power Letter, F. Saunders to P. Webster, “Bruce A Environmental Assessment Study Report”, NK21-CORR-00531-3839, December 7, 2005.

[5] CNSC Regulatory Document REGDOC-2.3.3, "Operating Performance: Integrated Safety Reviews", *DRAFT*, August 2014.

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

[7] CNSC Regulatory Document RD-360, “Life Extension of Nuclear Power Plants”, February 2008.

[8] Bruce Power Correspondence, “Bruce A Units 3 and 4 Refurbishment for Life Extension and Continued Operations: ISR Safety Factor Reports”, NK21-CORR-00531-05976, June 2, 2008.

[9] Bruce Power Correspondence, “Bruce A Units 3 and 4 Refurbishment for Life Extension and Continued Operation: ISR Safety Factor Reports 5, 6 and 7”, NK21-CORR-00531-06076, July 22, 2008.

[10] Bruce Power Correspondence, “Bruce A Units 3 and 4 Refurbishment for Life Extension and Continued Operation: ISR Safety Factor Reports 1, 2, 3 and 4”, NK21-CORR-00531-06596, December 18, 2008.

[11] Bruce Power Report, “Bruce NGS A Units 3 and 4 Global Assessment Report and Integrated Implementation Plan”, NK21-REP-03600-00025 Rev 001, May 29, 2009.

[12] CNSC Letter, L. Levert to F. Saunders, “Record of Proceedings – Bruce Power Inc.”, NK21-CORR-00531-04349, July 5, 2006.

[13] CNSC Regulatory Standard RD/GD-98, “Reliability Programs for Nuclear Power Plants”, June 2012.

[14] CNSC Information Document INFO-0795, “Licensing Basis Objective and Definition”, January 2010.

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

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

[17] Letter, F. Saunders to P. Elder, “Application for the Renewal of the Bruce A Power Reactor Operating Licence”, July 31, 2008, NK21-CORR-00531-06234.

[18] Letter, F. Saunders to M. Leblanc, “Supplementary Information Supporting the Bruce A and Bruce B Licence Renewal Applications”, October 22, 2008, NK21-CORR-00531-06486 and NK29-CORR-00531-07724.

[19] Letter, F. Saunders to M. Leblanc and K. Lafreniere, “Supplementary Information Supporting the Bruce A Licence Renewal Application and Information to Allow Fuelling and Restart of Bruce A Units 1 and 2”, June 30, 2009, NK21-CORR-00531-07031.

[20] Letter, F. Saunders to M. Leblanc, “Bruce A and Bruce B: Supplementary Information for Hearing Day 2”, July 21, 2009, NK21-CORR-00531-07162 and NK29-CORR-00531-08307.

[21] Letter, F. Saunders to K. Lafreniere, “Extension of Commitments Made in the Bruce A and Bruce B PROL Renewal Applications”, April 1, 2009, NK21-CORR-00531-06917.

[22] Bruce Power Report, “Review of Codes and Standards for the 2013/2014 Periodic Safety Review”, B-REP-00701-12APR2013-050, April 2013

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

[24] “ANSI/ASME Code Reconciliation for Replacement Material, Parts, and Components”, Revision 8, Reedy Engineering, May 21, 2012.

[25] Bruce Power Report, “Risk Informed Decision Making Process”, B-REP-03611-00004, Rev 000, October 31, 2008.

[26] “Management System Manual”, BP-MSM-1, Rev 012, June 23, 2014.

[27] Application for the Renewal of the Power Reactor Operating Licence for Bruce Nuclear Generating Station A, NK21-CORR-00531-10873, 31 October 2013.

Appendix A – Safety Factor Review Tasks

IAEA Specific Safety Guide, SSG-25, “Periodic Safety Review for Nuclear Power Plants” [2] provides guidance on the scope of the review. The guide breaks down the safety design, operation and management of an NPP into 14 Safety Factors. Each Safety Factor covers a number of review tasks so that the ISR is a comprehensive safety review.

One of the steps in preparing the ISR 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 ISR.

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 additional tasks for the Radiation Protection Safety Factor.

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 ISR(s). The review of plant design (including site characteristics) includes the following tasks:

1. Review of the list of SSCs important to safety for completeness and adequacy.
2. Review to verify that design and other characteristics are appropriate to meet the requirements for plant safety and performance for all plant conditions and the applicable period of operation, including:

    a. The prevention and mitigation of events (faults and hazards) that could jeopardize safety;

    b. The application of defence in depth and engineered barriers for preventing the dispersion of radioactive material (integrity of fuel, cooling circuit and containment building);

    c. Safety requirements (for example, on the dependability, robustness and capability of SSCs important to safety); and

    d. Design codes and standards.
3. Identification of differences between standards met by the nuclear power plant's design (for example, the standards and criteria in force when it was built) and modern nuclear safety and design standards;
4. Review of the adequacy of the design basis documentation;
5. Review for compliance with plant design specifications;
6. Review of the safety analysis report or licensing basis documents following plant modifications and in light of their cumulative effects and updates to the site characterization;
7. Review of plant SSCs important to safety to ensure that they have appropriate design characteristics and are arranged and segregated in such a way as to meet modern requirements for plant safety and performance, including the prevention and mitigation of events that could jeopardize safety; and
8. Review of the strategy for the spent fuel storage and conduct of an engineering assessment of the condition of the storage facilities, the records management and the inspection regimes being used.

A.1 Plant Safety Factors

SF2 Actual Condition of SSC's – 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.

A.1 Plant Safety Factors

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 ISR. 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.

A.1 Plant Safety Factors

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 ISR; 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.

A.2 Safety Analysis Safety Factors

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

A.2 Safety Analysis Safety Factors

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 ISR, 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 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).

b. 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.).

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 any 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.

A.3 Safety Performance and OPEX and Research & Development Safety Factors

SF9 External OPEX and Research & Development – 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.

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.

A.4 Management Safety Factors

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.

A.4 Management Safety Factors

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.

A.4 Management Safety Factors

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 whether the plant design has an adequate design for minimizing doses, both to workers and to the public.

The review tasks are as follows:

Verification whether the radiation protection program is appropriate and sufficiently comprehensive. In particular, the review should verify that radiation exposures are managed and controlled so that the deterministic effects of radiation exposure are prevented and the stochastic effects are reduced to the extent reasonably achievable.

Additionally:

a. The applicable requirements for the control of important sources of radiation and contamination have been addressed in the design and are sufficient to ensure that occupational doses remain within established limits and are as low as reasonably achievable;

b. Potential new sources of radiological impact have been recognized and addressed by the operating organization;

c. Methods to monitor radiation levels, contamination levels and occupational doses are adequate and corrective actions are taken to keep such levels with established limits and as low as reasonably achievable;

d. Sampling and measurement methods are consistent with current standards;

e. Records of radiation levels, contamination levels and occupational doses are being monitored and trended and appropriate actions are taken to keep such levels within established limits and as low as reasonably achievable;

f. Appropriate data on occupational doses, radiation levels and contamination levels have been provided to site personnel; and

g. The measures that have been implemented (including work planning, personal protective equipment and contamination control) are adequate to protect site personnel and prevent the spread of contamination when the plant is in normal operational states (including maintenance outages and anticipated operational occurrences).

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

The Bruce A Power Reactor Operating Licence – PROL 15.00/2015 [16] references regulatory documents, codes and standards that apply to the operation of the station. It also lists the key Bruce A station documents that support the operating licence. These Bruce Power documents, along with those listed in the licence renewal application submitted in 2008 [17] in support of the current operating licence, from supplementary submissions in 2008 [18] and 2009 [19], and from additional documents identified in the LCH as being part of the licensing basis [20][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 A the information is contained in the Bruce Power letter, "*Application for the Renewal of the Bruce A Power Reactor Operating Licence"*, July 31, 2008 [17] and in the subsequent submissions with supplementary information [18][19][20].

The majority of applicable Bruce Power documents are grouped in 12 attachments to the letter of application.

Index of Attachments

Attachment 1:	Site Description and Plan
Attachment 2:	Land Ownership and Control
Attachment 3:	Financial Guarantees
Attachment 4:	Evidence of Adequate Liability Insurance
Attachment 5:	Operational Support Documentation
Attachment 6:	Design and Operating Documentation
Attachment 7:	Results of the Program for Recruiting, Training and Qualifying Workers
Attachment 8:	Permits, Certificates and Licences and Summary of Environmental Regulatory Requirement
Attachment 9:	Station Improvement Plans
Attachment 10:	Safety Analysis Program
Attachment 11:	Information on Nuclear Substances
Attachment 12:	Action Levels Pertaining to Section 6 of the Radiation Protection Regulations

In addition to the information supporting the current licence, Table B-1 also lists the current Bruce Power governance documents included in the 2013 licence renewal application.

Table B-1: Bruce Power Governance Documents

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
B-CTP-35400-00001	Irradiated Fuel Shipping	R002	N	N	Y
B-HBK-09500-00003	Training – Performance Objectives and Criteria	R000	N	N	Y
B-HBK-09500-00005	Training – Performance Objectives and Criteria Evaluator Reference Material	R000	N	N	Y
B-REP-09034-00001	Bruce Power Reliability Program	R000	N	Y	Y
B-REP-31100-00003	Fuel Channel Condition Assessment	R003	N	N	Y
B-SMP-79100-00002	Operation Of the Tracker/Scanner	R002	N	Y	N
B-SMP-79100-00003	Operation Of The Waste Bag Monitor	R003	N	Y	N

[3] Current revision as of August 31, 2014.

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
B-SMP-79100-00004	Waste Processing	R007	N	Y	N
B-SMP-79100-00010	Operation Of TSA Barrel Monitor Models BM-286A And BM-286C	R004	N	Y	N
B-SMP-79100-00011	Waste Shielding Flask Use	R000	N	Y	N
B-SMP-79100-00012	High Activity Waste Cart Use	R001	N	Y	N
B-ST-03480-10000	Radionuclide effluent Monitoring System Requirements	R001	N	N	Y
BP-ERP-00001	Shift Emergency Controller Procedure	R018	N	Y	Y
BP-ERP-00042	Emergency Recovery Director	Superseded by BP-ERP-00061-R000 and BP-ERP-00062-R002	N	Y	Y
BP-MSM-1	Management System Manual	R012	Y	Y	Y
BP-NSAS-00016	Integrated Ageing Management for Safety Analysis	R000	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-OPP-00002	Operating Policies and Principles – Bruce A	R013	Y	Y	Y
BP-PLAN-00001	Bruce Power Nuclear Emergency Response Plan	R004	Y	Y	Y
BP-PLAN-00003	Bruce Power Electricity Emergency Plan	R005	N	N	Y
BP-PLAN-00004	Business Continuity Management	R008	N	N	Y
BP-PLAN-00005	Radioactive Material Transportation Emergency Response Plan	R005	N	Y	Y
BP-PLAN-00006	Conventional Emergency Plan	R001	N	Y	Y
BP-PLAN-00008	Fire Safety Management	R003	N	N	Y
B-PLAN-07292-00003	Bruce Nuclear Generating Station A – Spill Prevention Contingency	R005	N	N	Y
B-PLAN-07292-00004	Bruce Power Live Exercise and Spill Drill Planning – 5 Year Plan	R003	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
B-PLAN-31100-00001	Fuel Channel Life Cycle Management Plan	R005	N	N	Y
B-PLAN-33110-00001	Steam Generator and Preheater Life Cycle Management Plan	R004	N	N	Y
B-PLAN-33125-00001	PHT Feeder Piping Life Cycle Management Plan	R007	N	N	Y
BP-POLICY-01	Business Management	Obsolete	N	Y	N
BP-POLICY-02	Human Resource Management	Obsolete	N	Y	N
BP-POLICY-03	Information Resource Management	Obsolete	N	Y	N
BP-POLICY-04	Financial Resource Management	Obsolete	N	Y	N
BP-POLICY-05	Supply Chain, Facilities and Site Services Resource Management	Obsolete	N	Y	N
BP-POLICY-06	Regulatory Licensing	Obsolete	N	Y	N
BP-POLICY-07	Power Marketing	Obsolete	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-POLICY-08	Emergency Management & Security	Obsolete	N	Y	N
BP-POLICY-09	Long Term Corporate Planning Head of Strategy & Business Stakeholder Interaction	Obsolete	N	Y	N
BP-POLICY-09	Long Term Corporate Planning and Stakeholder Interaction	Obsolete	N	Y	N
BP-POLICY-10	Plant Design & Modification	Obsolete	N	Y	N
BP-POLICY-11	Plant Maintenance	Obsolete	N	Y	N
BP-POLICY-12	Plant Operation	Obsolete	N	Y	N
BP-POLICY-13	Legal	Obsolete	N	Y	N
BP-POLICY-14	Projects	Obsolete	N	Y	N
BP-POLICY-15	Nuclear Oversight	Obsolete	Y	Y	N
BP-PROC-00001	BMS Structural Change	R015	N	Y	N
BP-PROC-00003	Cobalt Handling	R002	N	Y	N
BP-PROC-00005	Limits to Hours of Work	R013	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00010	Emergency Preparedness Drills And Exercises	R008	N	Y	Y
BP-PROC-00011	Emergency Response Organization Staffing and Availability	R005	N	N	N
BP-PROC-00013	Human Factors Engineering Program Plan	R018	N	Y	N
BP-PROC-00014	Technical Operability Evaluation	R008	N	N	N
BP-PROC-00016	Business Assessment Process	R008	N	Y	N
BP-PROC-00019	Action Tracking	R002	N	Y	Y
BP-PROC-00020	Employee Temporary Assignment Process	R010	N	N	Y
BP-PROC-00024	Base Work Week for Management and Professional Staff	R004	N	Y	N
BP-PROC-00041	Contract Management	R022	N	N	N
BP-PROC-00052	Public Inquiry and Response	R006	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00058	CNSC Commitment Management	R010	N	Y	N
BP-PROC-00059	Event Response and Reporting	R021	N	Y	Y
BP-PROC-00060	Station Condition Record Process	R026	N	Y	Y
BP-PROC-00062	Processing External and Internal Operating Experience	R014	N	Y	Y
BP-PROC-00064	Formal Correspondence with the CNSC	R009	N	Y	Y
BP-PROC-00065	Station Condition Record Evaluation and Investigation	Superseded by BP-PROC-00518 & BP-PROC-00519	N	N	N
BP-PROC-00067	CNSC – Bruce Power Interface Protocol	R007	N	Y	N
BP-PROC-00068	Controlled Document Life Cycle Management	R021	N	Y	N
BP-PROC-00076	Management of the Off-Site Radiological Environmental Monitoring Program	R005	N	Y	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00079	MISA Regulation Requirements and Interpretation	R002	N	Y	Y
BP-PROC-00080	Monitoring of Radioactivity in Effluents	R006	N	Y	Y
BP-PROC-00090	Evaluation of Proposed Changed for CNSC Review and Approval	R004	N	Y	N
BP-PROC-00093	Spills to the Environment	R016	N	Y	Y
BP-PROC-00098	Records Management	R014	N	Y	Y
BP-PROC-00099	Conventional Emissions	R008	N	Y	Y
BP-PROC-00107	Waste Acceptance Criteria For Low And Intermediate Level Radioactive Waste	R005	N	Y	N
BP-PROC-00110	Information Classification, Access and Handling Requirements	R009	N	Y	N
BP-PROC-00127	Radioactive Liquid Emissions Response Procedure	R011	N	Y	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00133	Hazardous Waste Management Requirements	R005	N	Y	Y
BP-PROC-00137	Focus Area Self-Assessment	R014	N	Y	Y
BP-PROC-00140	Investment Proposal, Approval and Management	R008	N	Y	N
BP-PROC-00147	Benchmarking and Conference Activities	R015	N	Y	Y
BP-PROC-00150	Notifications Prior to Maintenance of Fire Systems	R002	N	N	Y
BP-PROC-00151	Job Filling and Reassignment for Society Represented Positions	R009	N	Y	Y
BP-PROC-00154	Cost Management Process	Obsolete	N	Y	N
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	Y
BP-PROC-00159	Control of Ignition Sources	R007	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00162	Business Risk Management – Business Risk Register	R006	N	N	N
BP-PROC-00163	Non-Licensed Operators Staffing of Days - Based Rotational Positions	R003	N	Y	Y
BP-PROC-00164	Authorized Nuclear Operator and Unit 0 Control Room Operator Staffing Days - Based Rotational Positions	R003	N	Y	Y
BP-PROC-00165	Reporting to Regulatory Agencies	R010	N	Y	N
BP-PROC-00166	General Procedure and Process Requirements	R023	N	Y	N
BP-PROC-00169	Safety-Related Systems List	R002	N	N	N
BP-PROC-00171	Radiological Emissions Limits and Action Levels	R017	N	Y	Y
BP-PROC-00174	Training – Administer Training Exemptions	R004	N	N	Y
BP-PROC-00175	Training- Prepare a Training Needs Analysis	R003	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00176	Training – Administer Remedial Training	R002	N	N	Y
BP-PROC-00181	Waste Minimization Plan	R004	N	Y	N
BP-PROC-00186	Fire Extinguishers	R002	N	N	Y
BP-PROC-00187	Fire Protection Impairment Control	R015	N	N	Y
BP-PROC-00188	Radioactive Material Transportation	R002	N	N	Y
BP-PROC-00189	Control of Transient Material	R011	N	N	Y
BP-PROC-00196	Landfill Waste Acceptance Criteria	R007	N	Y	Y
BP-PROC-00197	Chemistry Control Event Management	R005	N	N	N
BP-PROC-00199	Commissioning of Modifications	R007	N	Y	N
BP-PROC-00201	Training – Prepare Tests, Field Checkouts and Question & Answer Banks	R003	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00202	Training – Administer Tests and Field Checkouts	R002	N	N	Y
BP-PROC-00203	Training – Prepare a Job Analysis	R002	N	N	Y
BP-PROC-00204	Training – Perform a Task Analysis	R001	N	N	Y
BP-PROC-00205	Training – Prepare a Cost Analysis	Obsolete	N	N	Y
BP-PROC-00206	Training – Prepare Learning Objectives	R001	N	N	Y
BP-PROC-00207	Training – Prepare a Job Performance Measure	R002	N	N	Y
BP-PROC-00208	Training – Prepare Lesson Plans, Course Materials and Training Aids	R002	N	N	Y
BP-PROC-00209	Training - Administer Training Change Control	R007	N	N	Y
BP-PROC-00210	Training – Administer Vendor Training	R001	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00211	Training – Administer On-the-Job Training (OJT) and On-the-Job Evaluation (OJE)	R003	N	N	Y
BP-PROC-00212	Training – Administer Training Delivery	R004	N	N	Y
BP-PROC-00213	Training – Administer Training Evaluation	R004	N	N	Y
BP-PROC-00214	Training – Administer TIMS	R003	N	N	Y
BP-PROC-00215	Training- Administer Training Scheduling	R008	N	N	Y
BP-PROC-00216	Training – Prepare a Training and Qualification Description (TQD)	R006	N	N	Y
BP-PROC-00217	M&TE Calibration Program Requirements	R010	N	Y	Y
BP-PROC-00219	Source Segregation and Recycling Program	R007	N	Y	Y
BP-PROC-00221	Succession Management	R005	N	N	N
BP-PROC-00222	On-Boarding of Managers	R005	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00235	Fuel and Fuel Channel Integrated Safety Assessment	Superseded by BP-PROC-00893	N	N	N
BP-PROC-00237	Fuel and Fuel Channel Management Process	Superseded by BP-PROC-00893	N	N	N
BP-PROC-00238	Retention Process for Bruce Power Records	R011	N	Y	Y
BP-PROC-00244	Procurement Engineering	R004	N	N	N
BP-PROC-00246	Power Operated Valves	R003	N	N	N
BP-PROC-00247	Pressure Relief Devices	Superseded by BP-PROC-00078	N	N	N
BP-PROC-00248	Valve Packing	R003	N	N	N
BP-PROC-00249	Check Valves	R003	N	N	N
BP-PROC-00250	Writer's Guide for Station System Procedures	R007	N	Y	N
BP-PROC-00252	Control of Nonconforming Items	R009	N	Y	N
BP-PROC-00257	Procedure Use and Adherence	Superseded by BP-PROC-00617	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00259	Fire Protection for Relocatable Structures	R002	N	N	Y
BP-PROC-00261	Environmental Qualification	R005	N	Y	N
BP-PROC-00262	Warehouse Operations	R008	N	Y	Y
BP-PROC-00267	Management of Steam Generator and Preheater Tube Integrity	R000	N	N	N
BP-PROC-00268	Safety Related System Testing (SST) Program	R004	N	Y	Y
BP-PROC-00270	Bruce Power Tool Management Program	R007	N	Y	N
BP-PROC-00271	Observation and Coaching Procedure	R004	N	N	Y
BP-PROC-00274	External Independent Performance Assessment	R006	N	Y	N
BP-PROC-00276	Code of Conduct	R004	N	N	N
BP-PROC-00279	Joint Health and Safety Committee Operations	R004	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00284	Predictive Maintenance	R003	N	N	N
BP-PROC-00289	Fire Hose and Couplings	R001	N	N	Y
BP-PROC-00291	Work Program Analysis	Obsolete	N	Y	N
BP-PROC-00295	Planning and Scheduling Audits	R012	N	Y	N
BP-PROC-00299	Training – Administer Staff Capability	R003	N	N	Y
BP-PROC-00301	Reactivity Management	R002	N	N	Y
BP-PROC-00306	Chemical Risk Assessment Procedure	R011	N	N	Y
BP-PROC-00307	Control of Handling, Storage and Shipping	R014	N	N	Y
BP-PROC-00311	Life Cycle Management Plan	Obsolete	N	N	N
BP-PROC-00317	Crisis Management	R004	N	Y	Y
BP-PROC-00324	Nuclear Criticality Safety Management	R004	Y	Y	N
BP-PROC-00327	New Work Initiation	R006	N	Y	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00328	New Work Prioritization and Approval	R013	N	Y	Y
BP-PROC-00329	lnage Work Scoping and Scheduling	R014	N	Y	Y
BP-PROC-00334	Periodic Inspection	R003	N	N	Y
BP-PROC-00335	Design Management	R006	N	Y	Y
BP-PROC-00340	Emergency Work	R002	N	Y	N
BP-PROC-00341	Work Management Procedure Deviation	Obsolete	N	Y	N
BP-PROC-00342	Planned Outage Management	R005	N	Y	Y
BP-PROC-00343	Forced Outage Management	R005	N	Y	Y
BP-PROC-00344	Five Year Outage Plan	R000	N	N	Y
BP-PROC-00355	Hiring Process (Contractors)	R006	N	N	N
BP-PROC-00360	Training – Administer Critical Knowledge Retention	R001	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00361	In-Service Testing and Inspection to Satisfy CAN/CSA N287.7-08 Requirements	R001	N	N	N
BP-PROC-00363	Nuclear Safety Assessment	R003	N	Y	Y
BP-PROC-00369	Approved Suppliers List Management	R011	N	N	N
BP-PROC-00379	Employee Wellness	R006	N	N	N
BP-PROC-00382	Performance and Conditioning Monitoring	Superseded by BP-PROC-00781	N	Y	N
BP-PROC-00387	Plant Inspection	R000	N	N	N
BP-PROC-00389	Conventional Safety Programs	R002	N	N	Y
BP-PROC-00393	Valve Assessment Process	R000	N	N	N
BP-PROC-00397	Management Performance Self-Assessment	Obsolete	N	Y	N
BP-PROC-00400	Life Cycle Management of Critical SSCs	R002	N	Y	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00401	Emergency Public Information Plan	Superseded by BP-PROC-00402 "Duty Media Officer"	N	Y	N
BP-PROC-00402	Duty Media Officer	R004	N	Y	N
BP-PROC-00407	Site Morning Routine	Obsolete	N	Y	N
BP-PROC-00412	Trending, Analyzing, and Reporting of SCRs	R005	N	Y	Y
BP-PROC-00429	Budget Preparation Process	R003	N	Y	N
BP-PROC-00430	Internal Independent Performance Assessment	Obsolete	N	Y	N
BP-PROC-00433	Design Change Package	Obsolete	N	N	N
BP-PROC-00437	Work Verification	Obsolete	N	Y	N
BP-PROC-00438	Design Document Change	R002	N	Y	N
BP-PROC-00440	Business Metric Evaluation and Reporting	R004	N	Y	N
BP-PROC-00452	Core Management	R000	N	Y	Y
BP-PROC-00455	Fuel Procurement	R001	N	Y	Y

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BP-PROC-00457	Development and Approval of Predefined	R007	N	N	N
BP-PROC-00460	Fuel Handling	R002	N	Y	Y
BP-PROC-00465	Hiring Process (Regular Positions)	R009	N	Y	Y
BP-PROC-00468	Workforce Planning Process	R007	N	Y	Y
BP-PROC-00470	Configuration Management Program Oversight and Trending	R004	N	N	N
BP-PROC-00485	Business Planning	R003	N	Y	N
BP-PROC-00486	Strategic Planning	R003	N	Y	N
BP-PROC-00488	Disclosure Management	R003	N	Y	N
BP-PROC-00492	Corporate Branding Management	R003	N	Y	N
BP-PROC-00495	Configuration Management Oversight	Superseded by BP-PROC-00470	N	N	N
BP-PROC-00499	Bruce Power Nuclear Peer Group Process	Superseded by B-HBK-08130-00001	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00500	Control of Unsecured Equipment in Seismically Qualified Areas	R001	N	N	N
BP-PROC-00501	Integrated Preventive Maintenance Program	R005	N	Y	N
BP-PROC-00506	Effectiveness Reviews	R007	N	N	Y
BP-PROC-00509	Bruce A and B Quarterly Report of Performance Indicators	R003	N	N	N
BP-PROC-00510	Certification Training – Job Analysis for Certification Training Programs	R004	N	N	Y
BP-PROC-00511	Certification Training – Task Analysis for Certification Training Program	R004	N	N	Y
BP-PROC-00512	Certification Training – Training Design for Certification Training Programs	R003	N	N	Y

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BP-PROC-00513	Certification Training – Training Development for Certification Training Programs	R004	N	N	Y
BP-PROC-00517	Health Physics Response to Radiation Overexposure	R001	N	N	Y
BP-PROC-00518	Root Cause Investigation	R006	N	Y	Y
BP-PROC-00519	Apparent Cause Evaluation	R008	N	Y	N
BP-PROC-00532	Critical Spare Parts and Strategic Component Assessment	R002	N	Y	N
BP-PROC-00533	Obsolescence Management	R001	N	Y	N
BP-PROC-00534	Technical Basis Assessment	R002	N	N	N
BP-PROC-00535	Written Communication with Environmental Regulators	R001	N	N	Y
BP-PROC-00539	Design Change Package	R015	N	Y	Y
BP-PROC-00559	Station Plant Health Committee	R004	N	N	N

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BP-PROC-00562	Bruce Power Fundamentals Document and Process Requirements	R003	N	Y	N
BP-PROC-00565	Certification Training-Independence and Confidentiality Requirements for Development and Implementation of Initial Certification and Re-Certification Examinations	R011	N	N	Y
BP-PROC-00566	Certification Training – Standards and Methodology for Certification Training Process	R08	N	N	Y
BP-PROC-00567	Certification Training – Development and Administration of Diagnostic Simulator-Based Re-Certification	R013	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
BP-PROC-00568	Certification Training-Development and Administration of Comprehensive Simulator – Based Examinations for Initial Certification Training Programs	R012	N	N	Y
BP-PROC-00570	Certification Training – Development and Administration of Written Re-Certification Examinations and Examination Material for Certified Staff	R005	N	N	Y
BP-PROC-00571	Certification Training – Development and Administration of Comprehensive Simulator-Based Re-Certification Examinations for Certified Staff (CST)	R012	N	N	Y
BP-PROC-00572	Certification Training – Remedial Training for Certification Training Programs	R006	N	N	Y

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BP-PROC-00574	Certification Training- Filing and Retention of Certification Training Records	R007	N	N	Y
BP-PROC-00576	Certification Training – Conduct of Continuing Training and Re-Certification Testing	R006	N	N	Y
BP-PROC-00577	Certification Training – Conduct of Initial Certification Training	R002	N	N	Y
BP-PROC-00583	Ministry of Labour Interface Management	R001	N	N	Y
BP-PROC-00589	Outage Fundamentals	R003	N	N	Y
BP-PROC-00595	Training Fundamentals	R001	N	N	Y
BP-PROC-00596	Occupational Health and Safety Hazards and Applicable Legal Requirements	R004	N	N	Y
BP-PROC-00599	Engineering Guidance for Preventive Maintenance Program Support	R000	N	Y	N

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BP-PROC-00604	Training – Non-Licensed Operator Continuing Training	R006	N	N	Y
BP-PROC-00608	Organization Manual Requirements	Obsolete	N	Y	N
BP-PROC-00617	Human Performance Tools for Workers	R005	N	N	Y
BP-PROC-00619	Occupational Health and Safety Management Review	R002	N	N	Y
BP-PROC-00651	Safety Performance Metrics and Monitoring	R001	N	N	Y
BP-PROC-00653	Training – Non-licensed Operator Continuing Training	R003	N	N	Y
BP-PROC-00659	Severe Accident Management Procedure	R002	N	N	Y
BP-PROC-00667	Certification Training – Conduct of Continuing Training for Authorized/Responsible Health Physicists	R000	N	N	Y
BP-PROC-00695	Maintenance Program Basis	R003	N	N	Y

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BP-PROC-00696	Maintenance Organization	R002	N	N	Y
BP-PROC-00697	Maintenance Activities	R001	N	N	Y
BP-PROC-00698	Structures, Systems or Components (SSC) Monitoring	R001	N	N	Y
BP-PROC-00699	Maintenance Work	R003	N	N	Y
BP-PROC-00703	Change Management Guidance	R001	Y	N	N
BP-PROC-00707	Conventional Safety Instrumentation Management	R003	N	N	Y
BP-PROC-00722	Pandemic Response	R003	N	N	Y
BP-PROC-00734	Plant Status Control	R005	N	N	Y
BP-PROC-00752	Training – Prepare Computer Based Training	R003	N	N	Y
BP-PROC-00773	Hazardous Waste Management and Disposal Requirements	R001	N	N	Y
BP-PROC-00778	Scoping and Identification of Critical SSCs	R001	N	N	Y

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BP-PROC-00779	Continuing Equipment Reliability Improvement	R000	N	N	Y
BP-PROC-00780	PM Implementation	R001	N	N	Y
BP-PROC-00781	Performance Monitoring	R002	N	N	Y
BP-PROC-00782	Problem Identification and Resolution	R000	N	N	Y
BP-PROC-00783	Long-Term Planning and Life Cycle Management	R001	N	N	Y
BP-PROC-00794	Monitoring Human Performance	R002	N	N	Y
BP-PROC-00795	Human Performance Tools for Knowledge Workers	R000	N	N	Y
BP-PROC-00811	Procedure Alterations	R002	N	N	Y
BP-PROC-00833	Reporting to the CNSC	R000	N	N	Y
BP-PROC-00839	Reporting to the CSNC/IAEA Safeguards	R000	N	N	Y
BP-PROC-00842	Compressed Gas Storage	R002	N	N	Y

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BP-PROC-00848	Training Administer Manual Credits and Credit Corrections	R00	N	N	Y
BP-PROC-00857	Fire Barriers	R000	N	N	Y
BP-PROC-00870	Spill Management and Contaminated Lands Program	R002	N	N	Y
BP-PROC-00872	Conventional Safety Observations and Inspections	Obsolete	N	N	Y
BP-PROC-00878	Radioactive Waste Management	R000	N	N	Y
BP-PROC-00888	Conventional and Hazardous Waste Management	R000	N	N	Y
BP-PROC-00892	Nuclear Safety Culture Monitoring	R000	N	N	N
BP-PROC-00919	Stakeholder Information Disclosure	R000	N	N	Y
BP-PROG-00.01	Nuclear Safety Management	Obsolete	N	Y	Y
BP-PROG-00.02	Environmental Safety Management	R008	Y	Y	Y

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BP-PROG-00.03	Fire Safety Management	Superseded by BP-PLAN-00008	N	Y	N
BP-PROG-00.04	Pressure Boundary Quality Assurance Program	R020	Y	Y	Y
BP-PROG-00.06	Health and Safety Management	R008	Y	Y	Y
BP-PROG-00.07	Human Performance Program	R010	Y	Y	Y
BP-PROG-00.08	Corporate Risk Management	Superseded by BP-PROG-01.01	Y	Y	N
BP-PROG-01.01	Business Plan Management	R005	Y	Y	N
BP-PROG-01.02	Bruce Power Management System (BPMS) Management	R007	Y	Y	N
BP-PROG-01.03	Business Improvement Project Planning	Superseded by BP-PROG-14.01 Rev. 005	N	N	N
BP-PROG-01.04	Leadership Talent Management	R006	Y	Y	N
BP-PROG-01.05	Performance Monitoring, Assessment and Reporting	Obsolete	N	Y	N

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BP-PROG-01.06	Operating Experience Program	R014	Y	Y	Y
BP-PROG-01.07	Corrective Action	R010	Y	Y	Y
BP-PROG-01.08	Investment Management	Cancelled	N	Y	Y
BP-PROG-02.01	Worker Staffing	R009	Y	Y	Y
BP-PROG-02.02	Worker Learning and Qualification	R012	Y	Y	Y
BP-PROG-02.03	Worker Health and Safety Services	Obsolete	N	Y	N
BP-PROG-02.04	Worker Development and Performance Management	R010	Y	Y	N
BP-PROG-02.05	Worker and Employee Relations, Compensation and Benefits	superseded by BP-PROG-02.06 and BP-PROG-02.08	N	Y	N
BP-PROG-02.06	Worker/Labour Relations	R002	Y	Y	N
BP-PROG-02.07	Employee Communication	R000	Y	Y	N
BP-PROG-02.08	Resourcing and Total Rewards	R005	Y	Y	N

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BP-PROG-03.01	Document Management	R015	Y	Y	Y
BP-PROG-03.02	Information Technology	R004	Y	Y	N
BP-PROG-04.01	Financial Reporting and Control	R007	Y	Y	N
BP-PROG-05.01	Supply Chain	R013	Y	Y	Y
BP-PROG-05.03	Site Services	R009	Y	Y	Y
BP-PROG-06.01	CNSC Licence Acquisition	R004	Y	Y	N
BP-PROG-06.03	CNSC Interface Management	R003	Y	Y	Y
BP-PROG-07.01	Electricity Revenue Management	R008	Y	Y	N
BP-PROG-07.04	Scheduling and Dispatch of Plant	R004	Y	Y	N
BP-PROG-08.01	Emergency Management Program	R007	Y	Y	Y
BP-PROG-08.02	Nuclear Security	R006	Y	Y	Y
BP-PROG-09.02	Stakeholder Interaction	R005	Y	Y	Y

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BP-PROG- 10.01	Plant Design Basis Management	R008	Y	Y	Y
BP-PROG-10.02	Engineering Change Control	R009	Y	Y	Y
BP-PROG-10.03	Configuration Management	R005	Y	Y	N
BP-PROG-11.01	Equipment Reliability	R004	Y	Y	Y
BP-PROG-11.02	On-Line Work Management Program	R006	Y	Y	Y
BP-PROG-11.03	Outage Work Management	R005	Y	Y	Y
BP-PROG-11.04	Plant Maintenance	R006	Y	Y	Y
BP-PROG-12.01	Conduct of Plant Operations	R007	Y	Y	Y
BP-PROG-12.02	Chemistry Management	R005	Y	Y	Y
BP-PROG-12.03	Fuel Management	R003	Y	Y	Y
BP-PROG-12.05	Radiation Protection Program	R003	Y	Y	Y
BP-PROG-12.06	Radioactive Waste Management	R002	Y	Y	Y
BP-PROG-12.07	Heavy Water Management	R001	Y	Y	N

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BP-PROG-13.01	Corporate Governance and Legal Services	R003	Y	Y	N
BP-PROG-13.02	Legal Services. Advice and Representation	Obsolete	N	Y	N
BP-PROG-14.01	Project Management and Construction	R005	Y	Y	N
BP-PROG-14.02	Contractor Management	R005	Y	Y	N
BP-PROG-15.01	Nuclear Oversight Management	R004	Y	Y	Y
BP-RPP-00001	Radiation Protection Policies and Principles	R001	N	N	N
BP-RPP-00002	Radiation Protection Requirements	R001	N	Y	N
BP-RPP-00003	Radiation Protection Procedures Manual - General Information	Obsolete	N	Y	N
BP-RPP-00005	Routine Radiological Survey	R007	N	Y	N
BP-RPP-00006	Routine Protection Qualification	R000	N	Y	N

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BP-RPP-00007	Decontamination	R008	N	Y	N
BP-RPP-00008	Access Control	R005	N	Y	N
BP-RPP-00009	Dose Limits and Exposure Control	R008	N	Y	N
BP-RPP-00010	Segregation and Handling of Radioactive Waste	R005	N	Y	N
BP-RPP-00011	Radiation Protection Requirements for Work Planning	R011	N	Y	N
BP-RPP-00012	Use of Radiation Protection Instrumentation	R005	N	Y	N
BP-RPP-00013	Radioactive Shipments	R008	N	Y	N
BP-RPP-00014	Selection of Radiation Personal Protective Equipment	R007	N	Y	N
BP-RPP-00015	Zoning	R011	N	Y	N
BP-RPP-00016	Purchase, Use, Storage and Disposal of Licensed Sources	R004	N	Y	N

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BP-RPP-00018	Facility Access and Working Rights	R005	N	Y	N
BP-RPP-00019	Greenmanning, Protection Assistants and Access Desk Operation	R005	N	Y	N
BP-RPP-00020	Dosimetry and Dose Reporting	R012	N	Y	N
BP-RPP-00021	Use of Facility Change Rooms	R004	N	Y	N
BP-RPP-00022	Contamination Control	R010	N	Y	N
BP-RPP-00023	Hazard Surveys, Posting and Radiological Log	R010	N	Y	N
BP-RPP-00026	Designation of the Nuclear Energy Worker	R003	N	Y	N
BP-RPP-00027	Contaminated Tools and Equipment	R006	N	Y	N
BP-RPP-00033	Contractor Radiography Oversight	R004	N	Y	N
BP-RPP-00033	Unconditional Releases and Conditional Transfers of Material	R004	N	Y	N

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BP-SM-00010	Chemical Storage Cabinets	R013	N	N	Y
BP-SM-00045	Bruce Power Asbestos Control	R006	N	N	Y
BP-SM-00054	Hazardous Materials Approval Process	Superseded by BP-PROC-00306	N	N	Y
BP-SM-00075	Transportation of Dangerous Good Non-radioactive	R002	N	N	Y
BP-SM-00080	Workplace Hazardous Materials Information System	R001	N	N	Y
B-ST-07290-10000	MISA Compliance Technical Specifications	R000	N	Y	Y
DIV-EM-00007	Emergency Measure Program Assessment	Superseded by SEC-EPP-00007	N	N	Y
DIV-ENG-00001	PASSPORT Equipment Data Management	Superseded by BP-PROC-00584	N	N	N
DIV-ENG-00010	Probabilistic Risk Assessment Process	R000	N	N	Y
DIV-ENG-00012	Nuclear Safety Assessment Initiation and Review	R001	N	N	N

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DIV-ENG-00013	Planning of Internal Work	R001	N	N	N
DIV-ENG-00017	System and Item Classification	R001	Y	N	N
DIV-ENG-00018	Design Registration and Reconciliation	R001	Y	N	N
DIV-MTD-00017	Maintenance Work Execution	Superseded by BP-PROC-00685 and DPT-MPRS-00002	N	Y	N
DIV-OD-00028	Probabilistic Risk Assessment	Superseded by DIV-ENG-00010 Rev. 000	N	N	N
DIV-OPA-00001	Station Shift Complement – Bruce A	R010	Y	Y	Y
DIV-OPA-00003	Response to Transients – Bruce A	R007	N	N	N

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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	N	Y
DOM-BAOP-00001	Bruce A Operations – Division Organization Manual	R002	N	N	Y
DOM-CA-00001	Corporate Affairs	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-CROAD-00001	Corporate Risk Oversight and Audit – Division Organizational Manual	Obsolete	N	Y	N
DOM-CSBD-00001	Corporate Strategy and Business Development Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-DMES-00001	Engineering Support- Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-DMSE-00001	Station Engineering – Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y

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DOM-ENG-00001	Engineering – Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
DOM-FS-00001	Financial – Division Organizational Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	Y
DOM-FINCON-00001	Corporate Controller-Division Organizational Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-HR-00001	Human Resources and Communications – Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
DOM-IT-00001	IT Infrastructures and Operations	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-LAW-00001	Law Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-LR-00001	Labour Relations – Division Organizational Manual	Obsolete	N	Y	N
DOM-MKT-00001	Power Marketing	R010	N	N	Y
DOM-MS-00001	Nuclear Maintenance Services - Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N

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DOM-NORA-00001	Nuclear Oversight and Regulatory Affairs	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-NSI-00001	Nuclear Safety Integration – Division Organizational Manual	Obsolete	N	Y	N
DOM-NUCOS-00001	Nuclear Operations Support	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-OPA-00001	Bruce A Station – Division Organization Manual	R002	N	N	Y
DOM-OPA-00002	Bruce A Station – Division Organization Manual	R002	N	Y	N
DOM-OTG-00002	Outage Division – Division Organization Manual	R002	N	N	Y
DOM-PMCBMG-00001	PMC Business Management	R001	N	N	Y
DOM-PMCCON-00001	PMC Construction Division Organization Manual	Obsolete	N	N	Y
DOM-PMCPP-00001	Programs and Projects Division	Superseded by BP-MSM-1 Sheet 002	N	N	Y

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DOM-PFM-00001	Portfolio Management – Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-RO-00001	Regulatory Oversight – Division Organizational Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
DOM-SC-00001	Supply Chain – Division Organizational Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	Y
DOM-SEC-00001	Security – Division Organizational Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
DOM-SOS-00001	Safety and Operations Support – Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
DOM-SSER-00001	Site Services	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-TRG-00001	Training – Division Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	N	Y
DOM-WM-00001	Work Management and Outages - Division Organization Manual	Obsolete	N	Y	N

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DPM-PMO-00002	Project Management Office - Department Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
DPT-CHM-00003	Control of System Chemistry	R006	N	Y	Y
DPT-CHM-00006	Analytical Capability	R012	N	Y	Y
DPT-CHM-00008	Outage Chemistry Program	R005	N	N	N
DPT-ERO-00008	Emergency Services Team Drills And Exercises	R004	N	Y	Y
DPT-NSAS-00001	Quality Assurance of Safety Analysis	R005	N	N	N
DPT-NSAS-00002	Safety Report Analysis Update Process Overview	R004	N	N	Y
DPT-NSAS-00003	Guidelines for Evaluating and Prioritizing Safety Report Issues	R004	N	N	Y
DPT-NSAS-00004	Guidelines for Managing the Key Deliverables of the Safety Report Analysis Update Process	R004	N	N	N

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DPT-NSAS-00005	Guidelines for Controlling the Safety Report Basis	Superseded by DPT-NSAS-00004	N	N	N
DPT-NSAS-00006	Guidelines for Managing the Safety Report Issue Database	Superseded by DPT-NSAS-00004	N	N	N
DPT-NSAS-00007	Processing of S-99 Reportable Conditions Arising from Safety Analysis	R004	N	N	N
DPT-NSAS-00008	Management of External Work for Nuclear Safety Analysis and Support	R004	N	N	N
DPT-NSAS-00009	Assessment Guidelines Using PRA	Superseded by DIV-ENG-00010	N	N	N
DPT-NSAS-00010	Nuclear Safety Analysis Planning, Calculation and Verification	Superseded by DIV-ENG-00013	N	N	N
DPT-NSAS-00011	Configuration Management of Safety Analysis Software	R004	N	N	N
DPT-NSAS-00012	Preparation and Maintenance of Operational Safety Requirements	R003	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
DPT-NSAS-00013	Guidelines for Managing Reference Data Sets	R003	N	N	N
DPT-NSAS-00014	Requirements on Criticality Safety Evaluations	Obsolete	N	N	N
DPT-NSAS-00015	Execution of Safety Analysis	R004	N	N	N
DPT-PDE-00001	Human Factors Minor Change	R003	N	Y	N
DPT-PDE-00005	System and Item Classification	Superseded by DIV-ENG-00017	Y	N	N
DPT-PDE-00013	Bruce Power Human Factors Engineering Program Plan	R008	Y	N	N
DPT-PDE-00017	Bruce Power Seismic Qualification Standard	R005	N	N	N
DPT-PDE-00019	Steam Protection Barriers	R002	N	N	N
DPT-PDE-00027	Fire Hazard Assessment Preparation and Maintenance	R003	N	N	N
DPT-PDE-00028	Fire Safe Shutdown Analysis Maintenance	R004	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
DPT-PDE-00029	Fire Protection Code Compliance Review Maintenance	R003	N	N	N
DPT-PDE-00030	Fire Protection Technical Evaluations Preparation Review and Acceptance	R003	N	N	N
DPT-PDE-00031	Third Party Review - Fire Protection	R005	N	N	N
DPT-PE-00008	System and Component Performance Monitoring Plans	R005	N	N	N
DPT-PE-00009	System and Component Performance Monitoring Walkdowns	R000	N	N	N
DPT-PE-00010	System Health Reporting	R006	N	N	N
DPT-PE-00011	Component Program Health Reporting	R003	N	N	N
DPT-RS-00002	Risk Assessment of Operational Events	R000	N	N	N
DPT-RS-00003	Evaluation of Risk Outside the Scope of the Probabilistic Risk Assessment	R001	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
DPT-RS-00004	Risk Assessment of Proposed Changes to Engineering, Operations, Surveillance and Maintenance	R001	N	N	N
DPT-RS-00006	Outage and Inage Risk Management	R001	N	N	N
DPT-RS-00007	Preparation and Maintenance of Probabilistic Risk Assessments	R001	N	N	N
DPT-RS-00008	Preparation and Maintenance of Unavailability Models	R000	N	N	N
DPT-RS-00012	Systems Important to Safety (SIS) Decision Methodology	R001	N	N	N
EP-PROG-02.03	Worker Health and Safety Services	Obsolete	N	Y	N
GOM-CNOA-00001	Nuclear Operations Bruce A – Group Organization Manual	R000	N	N	Y
GOM-CORP-00001	Corporate Services Group Organization Manual	Obsolete	N	Y	N
GOM-HR-00001	Human Resources Group	Superseded by BP-MSM-1 Sheet 002	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
GOM-FIN-00001	Finance and Commercial Services Group Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	Y
GOM-NMS-00001	Nuclear Maintenance Services	Superseded by BP-MSM-1 Sheet 002	N	N	Y
GOM-NO-00001	Nuclear Oversight – Group Organization Manual	Obsolete	N	Y	N
GOM-OPS-00001	Nuclear Operations – Group Organization Manual	Superseded by BP-MSM-1 Sheet 002	N	Y	N
GOM-PMC-00001	Project Management and Construction Group Organization	Obsolete	N	N	Y
GOM-PRJT-00001	Projects – Group Organization Manual	Obsolete	N	Y	N
GRP-OPS-00003	Certification Training - Copilot Procedure	R014	N	Y	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
GRP-OPS-00005	Certification Training – Independence And Confidentiality For Development And Implementation Of Initial Certification And Re-Certification Examinations	Superseded by BP-PROC-00565	N	Y	N
GRP-OPS-00006	Certification Training - Standards And Methodology For Certification Training Progress Tests	Superseded by BP-PROC-00566	N	Y	N
GRP-OPS-00007	Certification Training - Development And Administration Of Diagnostic Simulator-Based Re-Certification Examinations For Certified Staff	Superseded by BP-PROC-00567	N	Y	N
GRP-OPS-00008	Certification Training - Requirements And Guidelines For Written And Oral Examinations For Initial Certification Training Programs	Superseded by BP-PROC-00568	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
GRP-OPS-00009	Certification Training – Development And Administration Of Comprehensive Simulator-Based Examinations For Initial Certification Training Programs	Superseded by BP-PROC-00569	N	Y	N
GRP-OPS-00010	Certification Training - Development And Administration Of Written Re-Certification Examinations And Examination Material For Certified Staff	Superseded by BP-PROC-00570	N	Y	N
GRP-OPS-00011	Certification Training - Development And Administration Of Comprehensive Simulator-Based Recertification Examinations (CST) For Certified Staff	Superseded by BP-PROC-00571	N	Y	N
GRP-OPS-00013	Certification Training – Filing Certification Training Records And Retention Of	Superseded by BP-PROC-00574	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
GRP-OPS-00014	Certification Training - Standards And Methodology For Conducting Remedial Training And Re-evaluation Of Candidates For CNSC Re-Certification	Superseded by BP-PROC-00572	N	Y	N
GRP-OPS-00017	Certification Training - Mentor Structured Learning	Superseded by BP-PROC-00575	N	Y	N
GRP-OPS-00025	Expectations of Duty Managers	R005	N	N	Y
GRP-OPS-00027	Certification Training - Conduct Of Continuing Training And Re-Certification Testing	Superseded by BP-PROC-00576	N	Y	N
GRP-OPS-00037	Certification Training - Conduct Of Initial Certification Training	Superseded by BP-PROC-00577	N	Y	N
GRP-OPS-00038	Bruce A and Bruce B Operations Standards and Expectation	R009	N	Y	Y
GRP-OPS-00047	Operator Routines and Inspections - Bruce A and Bruce B	R007	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
GRP-OPS-00055	Fitness for Duty Considerations for Shift Complement Staff Held Over	R001	N	N	Y
NK21-DM-XXX	There are 394 DMs listed in the licence renewal application and supplementary documentation, which are part of the licensing basis. Refer to [7] for details.	NA	N	Y	N
NK21-OM-XXX	There are 227 OMs listed in the licence renewal application and supplementary documentation, which are part of the licensing basis. Refer to [7] for details.	NA	N	Y	Y
NK21-OM-35370	Safeguards Operating Manual	R003	N	Y	N
NK21-OM-79500	Chemical Waste Management (Bruce A)	R006	N	N	Y
NK21-PIP-21100-00001	Periodic Inspection Program For Bruce NGS A Concrete Containment Structures And Appurtenances (Excluding Vacuum Building)	R002	N	Y	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
NK21-PIP-25100-00001	CSA N287.7-08 - Periodic Inspection Program for Bruce NGS A Vacuum Building	R001	N	Y	N
NK21-PLAN-03480-00001	Site Emission Monitoring Plans	R001	N	N	Y
NK21-REP-03482-00002	Derived Release Limits and Actions Levels for Bruce Nuclear Generating Station A	R002	N	N	Y
NK21-REP-03600-00021	Bruce NGS A Units 1 & 2 Global Assessment Report and Integrated Implementation Plan	R00	Y	N	N
NK21-REP-08160-00003	Bruce A Security Report	R007	Y	Y	N
NK29-REP-09071-00001	Description of The Bruce Power Nuclear Criticality Safety Program	R000	N	Y	N
NK21-SR-01320-00001	Part 1: Bruce A Safety Report - Plant and Site Description	R005	N	N	Y
NK21-SR-01320-00002	Part 2: Bruce A Safety Report – Plant Components and Systems	R005	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
NK21-SR-01320-00003	Part 3: Bruce A Safety Report – Accident Analysis	R004	N	Y	Y
NK37-SMP-79500-00001	Chemical Waste Transfer	R002	N	N	Y
NK37-SMP-79500-0002	Waste Oil Handling and Transfer	R001	N	N	Y
OM-WC-79500	Waste Chemical Transfer Facility (non-rad materials only)	R002	N	N	Y
PMC-XXX	There are 238 PMCs listed in the licence renewal application and supplementary documentation, which are part of the licensing basis. Refer to [7] for details.	NA	N	Y	N
PMC.6.3.004	Bruce A Return to Service Strategy	Obsolete	Y	N	N
SEC-CST-00001	General Field Guidelines at Bruce Learning Centre Fire Training Area	R005	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
SEC-CHD-00001	Guidelines for Preparing/Revising Chemistry Specifications	R001	N	Y	Y
SEC-DOCM-00035	Records Retrieval and Secure Storage	R014	N	Y	Y
SEC-EQD-00012	Start of Qualified Life	R002	N	N	N
SEC-EQD-00013	Environment Monitoring for EQ	R002	N	N	N
SEC-EQD-00021	Environmental Qualification Assessments	R003	N	N	N
SEC-EQD-00022	Development of Environmental Qualification Lists (EQL)	R003	N	N	N
SEC-EQD-00026	Environmental Qualification Qualified Life	Superseded by SEC-EQD-00021	N	N	N
SEC-EQP-00030	EQ Equipment Condition and Environment Monitoring Procedure	Superseded by SEC-EQD-00030	N	N	N
SEC-EQD-00031	Preparation of Environmental Qualification Dossiers (EQD)	R002	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
SEC-EQD-00032	Environmental Qualification Evaluations (EQE)	R005	N	N	N
SEC-EQD-00033	EQ Walkdown and Verification Process	R002	N	N	N
SEC-EQD-00034	Environmental Qualification Status Index	R003	N	N	N
SEC-FHO-00001	Safeguards Roles and Responsibilities	Obsolete	N	Y	N
SEC-ME-00007	Pipe Wall Thinning - FAC	1.1 Superseded by BP-PROC-00923 Rev. 000	N	N	N
SEC-ME-00008	Heat Exchangers	R001	N	N	N
SEC-MLIV-00011	Equipment Maintenance Strategy - Development, Approval and Oversight	Obsolete	N	N	N
SEC-MSS-00002	Maintenance Procedure Development and Revision	Superseded by BP-PROC-00694	N	Y	N
SEC-PE-00001	Item Equivalency Evaluation	R012	N	N	N

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
SEC-PE-00014	Selection of Shipping Storage and Handling Requirements	R004	N	N	Y
SEC-PSO-00005	Abnormal Incident Manual (AIM) Management	Superseded by BP-PROC-00250 Sheet 003	N	Y	N
SEC-PSO-00015	Requirements for Operating Procedures: Nuclear Units	Superseded by GRP-OPS-00050	N	Y	N
SEC-RPR-00022	Action Levels	R003	N	N	Y
SEC-SIMM-00001	Simulator Validation	R001	N	N	Y
SEC-SIMM-00002	Simulator Change Control	R001	N	N	Y
SEC-SSPE-00002	Maintenance of Air Conditioning and Refrigeration Equipment To Reduce Emissions of Refrigerants	R005	N	Y	Y
TQD-00009	Engineering Support Personnel	R008	N	N	Y
TQD-00012	Bruce A And Bruce B Authorized Nuclear Operator Initial Training And Qualifications Description	R005	N	Y	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
TQD-00013	Bruce A And Bruce B Control Room Shift Supervisor/Shift Manager Initial Training And Qualifications Description	R007	N	Y	Y
TQD-00014	Certified Staff Continuing Training And Recertification-Training And Qualification Description	R006	N	Y	Y
TQD-00015	Bruce A And Bruce B Certified Unit 0 Control Room Operator Initial Training And Qualification Description	R005	N	Y	Y
TQD-00019	Non-licensed Operators – Generating Units	R005	N	N	Y
TQD-00022	Control Maintenance	R007	N	N	Y
TQD-00023	Mechanical Maintenance	R005	N	N	Y
TQD-00030	Non-Licensed Operators Unit 0	R006	N	N	Y
TQD-00031	Non-Licensed Operators – General Services	R000	N	N	Y

Document Number	Document Title	Current Revision[3]	Referenced in Licence or LCH (Y/N)	Listed in Licence Renewal Application 2008 (Y/N)	Listed in Licence Renewal Application 2013 (Y/N)
TQD-00032	Non-Licensed Operators – Fuel Handling	R005	N	N	Y
TQD-00036	Chemical Technician	R008	N	N	Y
TQD-00046	Radiation Protection Technician	R002	N	N	Y
TQD-00075	Health Physicist	R003	N	N	Y
TQD-00039	Certification Section Instructor Training And Qualification Description	R010	N	Y	Y

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

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

a) the codes and standards that are referenced in the current Bruce A PROL or LCH;

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;

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 new Application for the Renewal of the Bruce A PROL [27]. Reference [27] identifies if there are transition plans for such codes and standards. Such transition plans could be used to address gaps in cases where a code or standard is included in the current licence, and has since been updated.

Table C-1: List of Codes and Standards

Document No.	Title	Revision in Licence or LCH[4]	Latest Revision Assessed[4]	Date of Latest Assessment[5]	Current Revision[6]	New Assessment Required[7]	Preliminary Assessment Type[8]	Relevant Safety Factor(s)[9]
CNSC G-149	Computer Programs Used in Design and Safety Analyses of Nuclear Power Plants and Research Reactors	N	NA	NA	2000/10	Y	HL	1, **5**
CNSC G-144	Trip Parameter Acceptance	N	NA	NA	2006/05	Y	HL	**5**
CNSC G-129	Keeping Radiation Exposures and Doses 'As Low As Reasonably Achievable (ALARA)'	Rev.1 2004/10	NA	NA	Rev. 1 2004/10	Y	HL	**15**
CNSC G-276	Human Factors Engineering Program Plans	N	NA	NA	2003/06	Y	HL	**12**
CNSC G-278	Human Factors Verification and Validation Plans	2003/06	2003/06	2006	2003/06	N	CV	12

[4] Revision noted in the current PROL (15.00/2014) or LCH. 'NA' denotes that this code or standard is not currently in the licence or LCH.

[5] Date of Latest Assessment: the date that the code or standards was last reviewed during one of the following previous activities:
- *Return to Service of Bruce A Units 3 and 4 (circa 2001);*
- *Life Extension of Units 1 and 2 (circa 2006);*
- *Proposed refurbishments of Units 3 and 4 Bruce B (circa 2008);*
- *Safety Basis & PSR for Units 1-8 (circa 2013) ;or*
- *No previous assessments completed (NA).*

[6] Current Revision as as of the CED of August 31, 2014.

[7] New assessment required (Y) when (1) Standard has been updated and/or (2) programs/implementation/analysis/design/ used in prior assessments have changed.

[8] Preliminary Assessment Type:
- *CBC: Clause by Clause;*
- *CV: Confirm Validity of Previous Assessment;*
- *HL: High-Level;*
- *CTC: Code to Code (Followed by HL or CBC;*
- *-: Regulatory documents and standards listed in PROL 15.00/2014 or LCH and that have not been revised will not be assessed as part of this ISR.*

[9] Relevant Safety Factor(s): **Lead** (i.e., which Safety Factor Report will Document the assessment) is underlined in bold only where a new assessment is identified as required.

Document No.	Title	Revision in Licence or LCH[4]	Latest Revision Assessed[4]	Date of Latest Assessment[5]	Current Revision[6]	New Assessment Required[7]	Preliminary Assessment Type[8]	Relevant Safety Factor(s)[9]
CNSC G-323	Ensuring the Presence of Sufficient Qualified Staff at Class I Nuclear Facilities – Minimum Staff Complement	2007/08	2007/08	2008	2007/08	N	CV	10, 12
CNSC R-10	The Use of Two Shutdown Systems in Reactors	N	1977	2013	1977/01	N	CV	1, 5, 8
CNSC R-77	Overpressure Protection Requirements for Primary Heat Transport Systems in CANDU Power Reactors Fitted with Two Shutdown Systems	N	1987/10	2013	1987/10	N	CV	1, 5, 7
CNSC R-116	Requirements for Leak Testing Selected Sealed Radiation Sources	N	NA	NA	1995/01	Y	HL	**15**
CNSC RD-204	Certification of Persons Working at Nuclear Power Plants	2008/02	NA	NA	2008/02	N	-	8, **10**, 12
CNSC RD-310	Safety Analysis For Nuclear Power Plants (Current: Deterministic Safety Analysis)	2008/02	2008/02	2013	CNSC REGDOC 2.4.1 (2014)	Y	CTC	**5**,7
CNSC RD-346	Site Evaluation for New Nuclear Power Plants	N	2008/11	2008	2008/11	N	CV	1, 7, 14
CNSC RD-353	Testing and Implementation of Emergency Measures	N	NA	NA	2008/11	Y	CBC	**13**
CNSC RD-360	Life Extension Of Nuclear Power Plants	2008/02	2008/02	2013	2008/02	N	-	All
CNSC RD/GD-99.3	Public Information and Disclosure	2013/03	NA	NA	RD/GD-99.3 (2012)	Y	CBC	8**,** 9,11, 13, 14
CNSC REGDOC 2.2.2	Personnel Training	N	NA	NA	2014/08	Y	CBC	8, 10, **12**
CNSC REGDOC 2.3.2	Accident Management Severe Accident Management Programs for Nuclear Reactors	N	2013	NA	CNSC REGDOC 2.3.2 (2013/09)	Y	CBC	5, **13**
CNSC REGDOC 2.5.2	Design of Reactor Facilities: Nuclear Power Plants	N	*Supersedes: RD-337*	2013	CNSC REGDOC 2.5.2 (2014)	Y	CTC	**1**, 5, 6, 7
CNSC REGDOC 2.6.3	Fitness for Service: Ageing Management	N	*Supersedes: RD-334*	2013	CNSC REGDOC 2.6.3 (2014)	Y	CTC	**4**

Document No.	Title	Revision in Licence or LCH[4]	Latest Revision Assessed[4]	Date of Latest Assessment[5]	Current Revision[6]	New Assessment Required[7]	Preliminary Assessment Type[8]	Relevant Safety Factor(s)[9]
CNSC REGDOC 2.9.1	Environmental Protection, Policies, Programs and Procedures at Class 1 Nuclear Facilities and Uranium Mines and Mills	N	*Supersedes: S-296 & G-296*	NA	CNSC REGDOC 2.9.1 (2013)	Y	CTC	11, **14**
CNSC S-98	Reliability Programs For Nuclear Power Plants	2005/07	2005/07	2013	RD/GD-98 (2012)	N	CV	2, 4, 6
CNSC S-99	Reporting Requirements for Operating Nuclear Power Plants	2003/03	2003/03	2006	CNSC REGDOC 3.1.1 (2014)	Y	CTC	8, 9, **10**, 11, 14
CNSC S-210	Maintenance Programs for Nuclear Power Plants	2007/07	2007/07	2013	RD/GD-210 (2012)	N	-	3, 4, 8,10,12
CNSC S-294	Probabilistic Safety Assessment For Nuclear Power Plants	2005/04	2005/04	2013	CNSC REGDOC 2.4.2 (2014)	Y	CTC	1, **6**
CNSC S-296	Environmental Protection, Policies, Programs And Procedures At Class I Nuclear Facilities And Uranium Mines And Mills	2006/03	2006/03	2008	CNSC REGDOC 2.9.1 (2013)	Y	CTC	11, **14**
CNSC Internal Guidance	CNSC Expectations for Licensee Hours of Work Limits - Objectives and Criteria	2010/08	2010/08	NA	NA	N	-	8, 10, 12
CNSC Internal Guidance	Requirements for the Requalification Testing of Certified Shift Personnel at Nuclear Power Plants	2009/05	2009/05	NA	NA	N	-	8, 10, 12
Examination Guide EG-1	Requirements and Guidelines for Written and Oral Certification Examinations for Shift Personnel at Nuclear Power Plants	2005/07	2005/07	NA	NA	N	-	8, 10, 12
Examination Guide EG-2	Requirements and Guidelines for Simulator-Based Certification Examinations for Shift Personnel at Nuclear Power Plants	2005/07	2004/06	NA	NA	N	-	8, 10, 12
CSA B51	Boiler, Pressure Vessel, and Pressure Piping Code	N	CSA B51-03 (2009)	2008	CSA B51-14 (2014)	Y	CTC	**1**, 2

Document No.	Title	Revision in Licence or LCH[4]	Latest Revision Assessed[4]	Date of Latest Assessment[5]	Current Revision[6]	New Assessment Required[7]	Preliminary Assessment Type[8]	Relevant Safety Factor(s)[9]
CSA N285.0	General Requirements For Pressure-Retaining Systems And Components In Candu Nuclear Power Plants	Update 1 2008	2008 Update No.1	2013	CSA N285.0-12 (2012) Update 1 (2013/09)	Y	CTC	**1**
CSA N285.2	Requirements for Class 1C, 2C, and 3C Pressure-Retaining Components and Supports in CANDU Nuclear Power Plants	N	NA	NA	CSA N285.2-99 (2014)	Y	HL	**1**
CSA N285.4	Periodic Inspection of CANDU Nuclear Power Plant Components	2005	2005	2008	CSA N285.4-14 (2014)	Y	CTC	**4**
CSA N285.5	Periodic Inspection of CANDU Nuclear Power Plant Containment Components	2008	CSA N285.5-08	2013	CSA N285.5-13 (2013)	Y	CV	**4**
CSA N285.8	Technical Requirements for In-Service Evaluation of Zirconium Alloy Pressure Tubes in CANDU Reactors	N	CSA N285.8-10 (2010)	2013	CSA N285.8-10 (2010)	N	CV	4
CSA N286-05	Management System Requirements for Nuclear Power Plants	Update 1 2007/11	CSA N286-05 Update 1 2007/11	2006	CSA N286-12	Y	CTC	1, 4, 5, 8, 9,**10**, 11, 12, 13, 14
CSA N286.7	Quality Assurance Of Analytical, Scientific And Design Computer Programs For Nuclear Power Plants	1999	1999	2008	CSA N286.7-99 (2012)	N	CV	1, 2, **5**, 6, 7
CSA N287.1	General Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N	CSA N287.1-M93 (1993)	2001	CSA N287.1-14 (2014)	Y	CTC	**1**
CSA N287.2	Material Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N	CSA N287.2-08 (2013)	2013	CSA N287.2-08 (2013)	N	CV	1
CSA N287.3	Design Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N	CSA N287.3-M93 (1993)	2001	CSA N287.3-14 (2014)	Y	CTC	**1**
CSA N287.4	Construction, Fabrication, and Installation requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	N	CSA N287.4-09 (2009)	2013	CSA N287.4-09 (2009)	N	CV	1

Document No.	Title	Revision in Licence or LCH[4]	Latest Revision Assessed[4]	Date of Latest Assessment[5]	Current Revision[6]	New Assessment Required[7]	Preliminary Assessment Type[8]	Relevant Safety Factor(s)[9]
CSA N287.5	Examination and Testing Requirements for Concrete Containment Structures for Nuclear Power Plants	N	CSA N287.5-11 (2011)	2013	CSA N287.5-11 (2011)	N	CV	1
CSA N287.7	In-Service Examination and Testing Requirements for Concrete Containment Structures for CANDU Nuclear Power Plants	2008	CSA N287.7-08 Update 1 (2013/09)	2013	CSA N287.7-08 (2013) Update 1 (2013/09)	N	CV	4
CSA N288.1	Guidelines for Calculating Derived Release Limits for Radioactive Material in Airborne and Liquid Effluents for Normal Operation of Nuclear Facilities	N	CSA N288.1-08 (2008)	2008	CSA N288.1-14 (2014)	Y	CTC	**14**
CSA N288.2	Guidelines for Calculating Radiation Doses to the Public from a Release of Airborne Radioactive Material under Hypothetical Accident Conditions in Nuclear Reactors	N	NA	NA	CSA N288.2-91 (1991)	Y	CV	**5,** 13
CSA N288.3.2	High Efficiency Air-Cleaning Assemblies for Normal Operation of Nuclear Facilities	N	CSA N288.3.2-M85 (1998)	2001	WITHDRAWN	NA	NA	1
CSA N288.4	Environmental Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N	CSA N288.4-M90 (1990)	2008	CSA N288.4-10 (2010)	Y	CBC	1, **14**
CSA N288.5	Effluent Monitoring Programs at Class I Nuclear Facilities and Uranium Mines and Mills	N	NA	NA	CSA N288.5-11 (2011)	Y	CBC	**14**
CSA N288.6	Environmental Risk Assessments at Class I Nuclear Facilities and Uranium Mines and Mills	N	NA	NA	CSA N288.6-12 (2012)	Y	CBC	6, **14**
CSA N289.1	General Requirements for Seismic Design and Qualification of CANDU Nuclear Power Plants	N	CSA N298.1-08 (2013)	2013	CSA N298.1-08 (2013)	N	CV	1, 3
CSA N289.2	Ground Motion Determination for Seismic Qualification of Nuclear Power Plants	N	N	N	CSA N289.2-10 (2010)	Y	HL	1, **3**
CSA N289.3	Design Procedures for Seismic Qualification of CANDU Nuclear Power Plants	N	CSA N289.3-M61 (2008)	2008	CSA N289.3-10 (2010)	Y	CTC	1, **3**

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CSA N289.4	Testing Procedures for Seismic Qualification of CANDU Nuclear Power Plants	N	CSA N289.4-M86 (2008)	2008	CSA N289.4-12 (2012)	Y	CTC	1, **3**
CSA N289.5	Seismic Instrumentation Requirements for CANDU Nuclear Power Plants	N	CSA N289.5-M91 (1991)	2006	CSA N289.5-12 (2012)	Y	CTC	**1**, 3
CSA N290.1	Requirements for the Shutdown Systems of CANDU Nuclear Power Plants	N	CSA N290.1-M80 (2001)	2006	CSA N290.1-13 (2013)	Y	CTC	**1**, 5
CSA N290.4	Requirements for Reactor Control Systems of Nuclear Power Plants	N	CSA N290.4-M82 (1982)	2013	CSA N290.4-11 (2011)	N	CV	1, 5
CSA N290.5	Requirements for Electrical Power and Instrument Air Systems of CANDU Nuclear Power Plants	N	CSA N290.5-06 (2006)	2013	CSA N290.5-06 (2011)	N	CV	1, 5
CSA N290.6	Requirements for Monitoring and Display of Nuclear Power Plant Safety Functions in the Event of an Accident	N	CSA N290.6-09 (2009)	2013	CSA N290.6-09 (2014)	N	CV	1, 5
CSA N290.13	Environmental Qualification Of Equipment For Candu Nuclear Power Plants	2005	CSA N290.13-05 (2005)	2008	CSA N290.13-05 (2013)	N	CV	1, 2, 3, 4, 5, 7
CSA N290.15	Requirements for the Safe Operating Envelope of Nuclear Power Plants	2010	NA	NA	CSA N290.15-10 (2010)	N	-	1, 2, **5**, 6, 7, 8
CSA N291	Requirements for Safety-Related Structures for CANDU Nuclear Power Plants	N	NA	NA	CSA N291-08 (2013)	Y	CBC	**1**, 2, 4
CSA N292.3	Management of Low- and Intermediate-Level Radioactive Waste	2008	NA	NA	CSA N292.3-14 (2014)	Y	CBC	11, **14**
CSA N293	Fire Protection For CANDU Nuclear Power Plants	2007	CSA N293-07	2008	CSA N293-12 (2012)	Y	CTC	**1**, 7
CSA N1600	General Requirements for Nuclear Emergency Management Programs	N	NA	NA	CSA N1600-14 (2014)	Y	CBC	**13**
CSA Z731	Emergency Planning for Industry	N	CSA Z731-03	2008	CSA Z731-03 (2014)	N	CV	13

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Darlington DG-38-03650-1	Purpose and Application of Nuclear Safety Design Guides	N	NA	2006	NA	N	-	1
Darlington DG-38-03650-2A	Common Mode Incidents – Overview and Design Requirements	N	NA	2006	NA	N	-	1, 7
Darlington DG-38-03650-2B	Common Mode Incidents – Seismic Design	N	NA	2006	NA	N	-	1, 7
Darlington DG-38-03650-3	Limiting Consequential Damage of Postulated Pipe Ruptures	N	NA	2006	NA	N	-	1, 7
Darlington DG-38-03650-4	Shutdown Systems	N	NA	2006	NA	N	-	1
Darlington DG-38-03650-5	Emergency Coolant Injection	N	NA	2006	NA	N	-	1
Darlington DG-38-03650-6	Containment	N	NA	2006	NA	N	-	1
Darlington DG-38-03650-7	Extensions of the Containment Envelope	N	NA	2006	NA	N	-	1
Darlington DG-38-03650-8	Environmental Qualification of Safety Related Equipment	N	NA	2006	NA	N	-	1, 3
Darlington DG-38-03650-9	Safety Assessments	N	NA	2006	NA	N	-	1
IAEA NS-G-3.2	Dispersion of Radioactive Material in Air and Water and Consideration of Population Distribution in Site Evaluation for Nuclear Power Plants	N	N	N	2002/03	Y	HL	**14**, 15
IAEA SSR-2/2	Safety of Nuclear Power Plants: Commissioning and Operation Specific Safety Requirements	N	NA	NA	SSR-2/2 (2011)	Y	CBC	**11**
IAEA SSG-25	Periodic Safety Review for Nuclear Power Plants	N	SSG-25 (2013)	2013	SSG-25 (2013)	N	-	All

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INPO 91-014	Guidelines for Radiological Protection at Nuclear Power Stations	N	NA	NA	INPO 91-014 Rev.1 1995/10	Y	HL	**15**
NBC (2010)	National Building Code of Canada	N	NBC (2005)	2013	NBC (2010) First Revisions and Errata (2012/12)	Y	CTC	**1**
NFC (2010)	National Fire Code	N	NFC (2005)	2013	NFC (2010)	Y	CTC	**1**
NFPA-801 (2011 Errata sheets issued 2011/5/2)	Standard for Fire Protection for Facilities Handling Radioactive Materials, Section 5 (Construction)	N	NFPA-801 (2011 Errata)	2008	NFPA-801 (2014)	Y	CTC	**1**
WANO GL 2004-01	Guidelines for Radiological Protection at Nuclear Power Stations	N	NA	NA	WANO GL 2004-01	Y	CBC	**15**