Canadian Nuclear Safety Commission | Commission canadienne de sûreté nucléaire

Specific Amendments for Fukushima Omnibus Amendment Project

To address

CNSC Fukushima Task Force Report

Combined Amendment and Rationale Tables for

S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*

S-296 (and G-296), *Developing Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills*

G-306, *Severe Accident Management Programs for Nuclear Reactors*

RD-308, *Deterministic Safety Analysis for Small Reactor Facilities*

RD-310, *Safety Analysis for Nuclear Power Plants*

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Canada

Preface

The CNSC regulatory document amendments proposed and discussed here were assembled under the mandate of the CNSC Fukushima Omnibus Amendment Project. These amendments address specific improvements and clarifications of regulatory requirements, as identified in the *CNSC Fukushima Task Force (FTF) Report* and the corresponding *CNSC Staff Action Plan on the CNSC Task Force Recommendations*.

The content of the tables in this document is intended to communicate the rationales for these focused changes to specific regulatory documents, and will be used to confirm the adequacy of these amendments. Following the consultation process, these amendments will be integrated into the existing regulatory documents for re-publication. The documents will be re-issued under the latest CNSC naming, numbering and nomenclature conventions.

Updated prefaces provide administrative history, links to the *CNSC Fukushima Task Force Report*, and explanations of the mandatory language used in CNSC regulatory and guidance documents.

References:

1. *CNSC Fukushima Task Force Report* (INFO-0824)
2. *CNSC Staff Action Plan on the CNSC Task Force Recommendations* (INFO-0828)

Red text = new text provided to address the Fukushima Task Force recommendations

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Part A: S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*

S-294 sets out the requirements for performance of Probabilistic Safety Assessments (PSA) by licensees.

It was identified that S-294 needed to be revised at the earliest opportunity to provide additional requirements and guidance required for upcoming refurbishments and new build.

The updated criteria to address lessons learned from the Fukushima event include:

- A Level 1 and 2 PSA should be required to cover irradiated fuel bay events and multi-unit considerations, as well as plant-wide internal fires, internal floods, seismic events and other external events.
- Some of the existing requirements should be made more specific. For example:
    - The PSA methodology and computer codes are required to be accepted by CNSC staff and two IAEA procedures are mentioned for background. However, no purpose is provided for the acceptance, or the means by which it may be achieved.
    - Although it is expected that the PSA methodology will verify that the safety goals in RD-337 are met, this is not stated (but should be).
    - Although the PSA methodology is required to identify dominant contributors to risk, plant vulnerabilities and provide insights into the management of severe accidents it is not expressly stated (but should be).
    - The means by which sensitivity and uncertainty analysis are to be performed should also be made clearer.

A new requirement is provided for advance CNSC consultation and/or acceptance of the expected uses of the PSA, since this will influence the methodology and codes.

Table A. S-294 Proposed Amendments and Rationale

S-294 Section #	Current Text	Proposed Changes	Rationale
Preface	**NA**	**Preface** **This regulatory document sets out the requirements of the Canadian Nuclear Safety Commission (CNSC) with respect to the probabilistic safety assessment (PSA).** **When published, this document will amend/supersede S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants*. This document has been amended to clarify or add criteria reflecting lessons learned from the Fukushima nuclear event of March 2011. The amendments were made to address findings from INFO-0824, *CNSC Fukushima Task Force Report*, as applicable to S-294.** **---** This document may be used as part of the licensing basis for a regulated facility or activity, including when referenced in a licence, either directly or indirectly (through licensee reference documents). The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and thus establishes the basis for the CNSC's compliance program in respect of that regulated facility or activity. The licensing basis for a regulated facility or activity is a set of requirements and documents comprising:	To provide the administrative history of the amended document, the legal basis, and an explanation of the mandatory language in regulatory and guidance documents. The rationale for the amendment, as related to the *CNSC Fukushima Task Force Report,* is also provided.

S-294 Section #	Current Text	Proposed Changes	Rationale
		(i) the regulatory requirements set out in the applicable laws and regulations (ii) the conditions and the safety and control measures described in the facility's or activity's licence, along with 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 In this document, "shall" is used to express a requirement – i.e., a provision that a licensee or licence applicant is obliged to satisfy, in order to comply with the requirements of this regulatory document. "Should" is used to express guidance, or that which is advised. "May" is used to express an option, or that which is permissible within the limits of this regulatory document. "Can" is used to express possibility or capability. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee's responsibility to identify and comply with all applicable regulations and licence conditions.	
1.	**Purpose** The purpose of this Regulatory Standard, when incorporated into a licence to construct or operate a nuclear power plant (NPP) or other legally enforceable instrument, is to assure that the licensee conducts a	**Purpose** The purpose of this **regulatory document,** when incorporated into a licence to construct or operate a nuclear power plant (NPP) or other legally enforceable instrument, is to assure that the licensee conducts a "probabilistic safety assessment (PSA)" in accordance with defined	The terminology has changed from regulatory standard to regulatory document.

S-294 Section #	Current Text	Proposed Changes	Rationale
	"probabilistic safety assessment (PSA)" in accordance with defined requirements.	requirements.	
2.0	**Scope** This Regulatory Standard sets out the requirements for the PSA that a licensee who constructs or operates a NPP shall conduct, when required by the applicable licence or other legally enforceable instrument.	**Scope** This **regulatory document** sets out the requirements for the PSA that a licensee who constructs or operates a NPP shall conduct, when required by the applicable licence or other legally enforceable instrument.	The terminology has changed from regulatory standard to regulatory document.
4.0	**Background** The following International Atomic Energy Agency (IAEA) Safety Series documents provide general guidance for conducting quality PSAs: *1.* IAEA Safety Series No. 50-P-4, *Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 1); and* *2.* IAEA Safety Series No. 50-P-8, *Procedures for Conducting Probabilistic Safety Assessments of Nuclear Power Plants (Level 2), Accident Progression, Containment Analysis and Estimation of Accident Source Terms.*	**Background** The following International Atomic Energy Agency (IAEA) safety **standards** documents **or updated versions**, provide general guidance for conducting quality PSAs: **1 IAEA safety standard SSG-3, *Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants,* and** **2. IAEA safety standard SSG-4, *Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants***	The references in the original S-294 are outdated and superseded by new IAEA safety series. There is also a need to specify IAEA and international standards for the determination of the quality of the PSA. The updating of IAEA references will partly address the following related to the FTF recommendations: The PSA methodology and computer codes are required to be accepted by CNSC, and two IAEA procedures are mentioned for background. A purpose is provided for the acceptance, and the means by which it may be achieved.
5.0	**PSA Requirements** The licensee shall carry out the following activities:	The licensee shall carry out the following activities:	

S-294 Section #	Current Text	Proposed Changes	Rationale
5.1	Perform **a facility specific Level 2** PSA for each NPP in question.	Perform a **Level 1** and Level 2 PSA for each NPP. **Radioactive sources other than the reactor core, such as the irradiated fuel bay, shall be considered. Multi-unit impacts, if applicable, shall be included.** **The PSA shall include:** **1. a systematic analysis, to give confidence that the design will comply with the general safety objectives** **2. demonstration that a balanced design has been achieved** **3. confidence that small deviations in plant parameters that could give rise to severely abnormal plant behaviour ("cliff-edge effects") will be prevented;** **4. assessments of the probabilities of occurrence for severe core damage states, and assessments of the risks of major radioactive releases to the environment.** **5. site-specific assessments of the probabilities of occurrence, and the consequences of external hazards** **6. identification of plant vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents, or mitigate their consequences** **7. assessment of the adequacy of emergency procedures**	To explicitly specify: • Level 1 and Level 2 • scope of initiating events to be considered • radioactive sources to be considered • multi-unit effect This will address the following related to the FTF recommendations: A Level 1 and 2 PSA is required to cover irradiated fuel bay events and multi-unit considerations, as well as plant-wide internal fires, internal floods, seismic events and other external events. The purpose of the PSA is taken from IAEA SSG-3, and lists in a very clear manner the purpose for conducting a PSA, which will address the following related to the FTF recommendations: It is now expressly stated that the PSA methodology is required to identify dominant contributors to risk, plant vulnerabilities and provide insights into the management of severe accidents. It is expected that the PSA methodology will verify that the safety goals in design (RD-337) are met, and this is now stated.

S-294 Section #	Current Text	Proposed Changes	Rationale
		8. assessment of insights into the severe accident management program	
5.2	Establish and apply a formal quality assurance process for conducting a PSA, such as the Canadian Standards Association (CSA) Standard N286.2, *Design Quality Assurance for Nuclear Power Plants*;	Establish and apply a formal **management system or quality assurance program** for conducting a PSA, such as the Canadian Standards Association (CSA) **Standard N286-05, *Management system requirements for Nuclear Power Plants*. The computer codes used for the PSA models shall comply with** CSA **N286.7-99, *Quality Assurance of Analytical, Scientific and Design Computer Programs for Nuclear Power Plants.***	CSA N286.2 is withdrawn. CSA standard N286-05 supersedes N286.0 as well as the associated sub-tiers N286.1 through N286.6. It is also important to add the CSA standard N286.7-99 regarding the QA program for the computer codes, in order to ensure the codes used in developing PSAs comply with the CSA standard. The original S-294 does not explicitly call for compliance with N286.7. This will help address the following related to the FTF recommendations: A requirement for advance CNSC consultation and/or acceptance of the expected uses of the PSA is provided, since this will influence the methodology and codes.
5.3	**Ensure that** the PSA models reflect the plant as built and operated, as closely as reasonably achievable within the limitations of PSA technology and consistent with risk impact;	The PSA models reflect the plant as built and operated **(including multi-unit impacts),** as closely as reasonably achievable within the limitations of PSA technology, and consistent with the risk impact;	To clarify that multi-unit effects have to be considered.
5.4	Update the PSA models every three years or sooner if major changes occur in the facility;	Update the PSA models every **five** years or sooner if major changes occur in the facility.	To align the PSA update with the safety analysis report update in S-99/RD-99.1 and with licence renewal.

S-294 Section #	Current Text	Proposed Changes	Rationale
5.5	Ensure that the PSA models are developed using assumptions and data that are realistic and practical;	Ensure that the PSA models are developed using assumptions and data that are realistic and practical. **Supporting deterministic safety analysis shall be provided.**	To provide the supporting analysis for the specification of the success criteria, assumption etc.
5.6	**Ensure that** the level of detail of the PSA is consistent with the **NPP** testing and configuration management programs;	The level of detail of the PSA is consistent with the **facility** testing, **maintenance** and configuration management programs, **and with the intended uses of the PSA.**	To specify that the level of details of the PSA should also be consistent with the intended use of the PSA. This will help address in part the following related to the FTF recommendations: A requirement for advance CNSC consultation and/or acceptance of the expected uses of the PSA is provided, since this will influence the methodology and codes.
5.7	Seek CNSC acceptance of the methodology and computer codes to be used for the PSA;	Seek CNSC acceptance of the methodology and computer codes to be used for the PSA, **prior to using them for the purpose of this document**. • **The methodology shall state the intended PSA applications.** • **The methodology shall be suitable for the intended PSA applications.** • **The computer codes used for PSA and for the supporting deterministic safety analyses shall be developed, validated, and used in accordance with a quality assurance program that meets the requirements of CSA N286.7-99.**	This will help address the following related to the FTF recommendations: The PSA methodology and computer codes are required to be accepted by CNSC, and two IAEA procedures are mentioned for background. A purpose for the acceptance, and the means by which it may be achieved, are provided. A requirement for advance CNSC consultation and/or acceptance of the expected uses of the PSA is provided, since this will influence the methodology and codes. The purpose of these changes is to clarify the separation between the computer codes for developing the PSA models and the codes used for deterministic safety analyses to draw the success criteria.

S-294 Section #	Current Text	Proposed Changes	Rationale
5.8	Include both internal and external events[1] in the PSA; [1] For external events, the licensee may, with the agreement of “persons authorized” by the Commission, choose an alternative analysis method to conduct the assessment. In such cases, the external event may be excluded from the PSA.	**Include all potential site-specific initiating events and potential hazards, namely: (a) internal initiating events caused by random component failures and human error; (b) internal hazards (e.g., internal fires and floods, turbine missiles) and (c) external hazards, both natural (e.g., earthquakes, high winds, external floods) and human-induced, but non-malevolent (e.g., airplane crashes, accidents at nearby industrial facilities).** **Also, include potential combinations of external hazards. Examples include seismic, floods, or fire.** **The screening criteria of hazards shall be acceptable to the CNSC.** The licensee may, with the agreement of “persons authorized” by the Commission Tribunal, choose an alternative analysis method to conduct the assessment **of external events (internal hazards and external hazards).**	The requirement is made clearer. This will address the following related to the FTF recommendations: A Level 1 and 2 PSA is required to cover irradiated fuel bay events and multi-unit considerations, as well as plant wide internal fires, internal floods, seismic events and other external events.
5.9	Include both at power and shutdown states in the PSA; and	**Include all operational states of the NPP (full power, low power, and shutdown).**	This clause has been reworded to be more inclusive and high-level, in order to address potential new build designs.
5.10	Include sensitivity analysis, uncertainty analysis and importance measures in the PSA.	No change	This requirement should remain unchanged (high-level), while the means by which these analyses are to be performed will be specified in GD-294, since the treatment of uncertainty and sensitivity may differ for the Level 1 PSA, Level 2 PSA, and seismic PSA.

S-294 Section #	Current Text	Proposed Changes	Rationale
5.11		**The PSA results may be repeated and reaffirmed.**	To ensure PSA quality.
5.12		**Documentation** **The licensee shall provide comprehensive and detailed documentation of the PSA, including assumptions, methodology, simplifications and results. It should include significant contributors and vulnerabilities, which would support the regulatory review and assessment of the PSA.**	This will help address the following related to the FTF recommendations: It is now expressly stated that the PSA methodology is required to identify dominant contributors to risk, plant vulnerabilities and provide insights into the management of severe accidents.

Part B: S-296 and G-296, *Developing Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills*

S-296 sets out the environmental protection policies, programs and procedures that licensees shall implement when required by applicable licence conditions.

The updated criteria in S-296 and G-296 now:

- Refine the scope of environmental monitoring for extreme emergency situations.
- Review the provisions of, and enhance environmental monitoring instrumentation, to ensure it is adequately robust against severe situations.
- Review the environmental monitoring layouts of equipment provisions for adequacy against severe situations.
- Establish and reinforce criteria and guidelines for environmental monitoring in emergency situations.

Table B1. S-296 Proposed Amendments and Rationale

S-296 Section #	Current Text	Proposed Changes	Rationale
Preface		**Preface** **This regulatory document sets out the requirements of the Canadian Nuclear Safety Commission (CNSC) with respect to the environmental protection policies, programs and procedures.** **When published, this document will amend/supercede S-296, *Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills*. The document has been amended to clarify or add criteria reflecting lessons learned from the Fukushima nuclear event of March 2011. Amendments to the accompanying guidance document address findings from INFO-0824, *CNSC Fukushima Task Force Report*.** -- This document may be used as part of the licensing basis for nuclear facilities and regulated activities, including when referenced in a licence, either directly or indirectly (through licensee reference documents). The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and thus establishes the basis for the CNSC's compliance program in respect of that regulated facility or activity.	To provide the administrative history of the amended document, the legal basis, and an explanation of the mandatory language in regulatory and guidance documents. The rationale for the amendment, as related to the *CNSC Fukushima Task Force Report,* is also provided.

Preface (cont)		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 the safety and control measures described in the facility's or activity's licence, along with 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 In this document, “shall” is used to express a requirement – i.e., a provision that a licensee or licence applicant is obliged to satisfy, in order to comply with the requirements of this regulatory document. “Should” is used to express guidance, or that which is advised. “May” is used to express an option, or that which is permissible within the limits of this regulatory document. “Can” is used to express possibility or capability. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee’s responsibility to identify and comply with all applicable regulations and licence conditions.	

	Relevant text in the referenced ISO-14001 documentation in section 4.4.7 states. *4.4.7 Emergency preparedness and response* *The organization shall establish, implement and maintain a procedure(s) to identify potential emergency situations and potential accidents that can have an impact(s) on the environment and how it will respond to them… The organization shall periodically review and, where necessary, revise its emergency preparedness and response procedures, in particular, after the occurrence of accidents or emergency situations.*	No change.	Section 4.4.7 of ISO-14001 (which was adopted by direct reference in S-296) already contains adequate generic clauses for emergency preparedness and response. Documents ISO-14001 and G-296 use the term "emergency" without distinguishing or defining degrees such as non-severe, severe or extreme. CSA N286-05 (section 6.26) also provides a requirement for emergency preparedness. The text of G-296 is refined to clarify that it includes environmental monitoring instrumentation for emergency situations. This addresses the following related to the FTF recommendations: Refine the scope of environmental monitoring for extreme emergency situations.

Table B2. G-296 Proposed Amendments and Rationale

G-296 Section #	Current Text	Proposed Changes	Rationale
Preface		**Preface** **This regulatory document sets out the expectations and guidance of the Canadian Nuclear Safety Commission (CNSC) with respect to the environmental protection policies, programs and procedures.** **When published, this document will amend/supercede G-296, *Developing Environmental Protection Policies, Programs and Procedures at Class I Nuclear Facilities and Uranium Mines and Mills*. This document has been amended to clarify or add criteria reflecting lessons learned from the Fukushima nuclear event of March 2011. The amendments were made to address findings from INFO-0824, *CNSC Fukushima Task Force Report*, as applicable to G-296.** --- This document may be used as part of the licensing basis for nuclear facilities and regulated activities, including when referenced in a licence, either directly or indirectly (through licensee reference documents). The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and thus establishes the basis for the CNSC's compliance program in respect of that regulated facility or activity.	To provide the administrative history of the amended document, the legal basis, and an explanation of the mandatory language in regulatory and guidance documents. The rationale for the amendment, as related to the *CNSC Fukushima Task Force Report,* is also provided.

Preface (cont)		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 the safety and control measures described in the facility's or activity's licence, along with 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 In this document, "shall" is used to express a requirement – i.e., a provision that a licensee or licence applicant is obliged to satisfy, in order to comply with the requirements of this regulatory document. "Should" is used to express guidance, or that which is advised. "May" is used to express an option, or that which is permissible within the limits of this regulatory document. "Can" is used to express possibility or capability. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee's responsibility to identify and comply with all applicable regulations and licence conditions.	

5.3.3	**Other Considerations** As a further consideration, the EMS should address environmental emergency preparedness and response in terms of 1. the proposed measures to prevent or mitigate the effects of accidental releases of nuclear substances and hazardous substances on the environment; and 2. the health and safety of persons.[27][28]	**Other Considerations** As a further consideration, the EMS should address environmental emergency preparedness and response in terms of: 1. the proposed measures to prevent or mitigate the effects of accidental releases of nuclear substances and hazardous substances on the environment **2. the proposed measures to ensure the availability and accessibility of environmental monitoring instrumentation during emergency situations** **3. the inclusion of environmental monitoring instrumentation and equipment layouts in emergency plans** ~~2.~~**4**. the health and safety of persons [27][28]	The guidance provided in ISO-14001, as quoted in S-296 on environmental monitoring for "*emergency situations and potential accidents*", is minimal. This indicates the need to provide some "lessons learned" guidance in section 5.3.3 of G-296 (accompanying S-296), related to the task force's recommendations. Specifically, this addresses the following related to the FTF recommendations: Establish and reinforce criteria and guidelines for environmental monitoring in emergency situations. Review the provisions of and enhance environmental monitoring instrumentation to ensure it is adequately robust against severe situations. Review the environmental monitoring layouts of equipment provisions for adequacy against severe situations. The CNSC is applying international guidance for environmental monitoring for emergency situations, and will continue to do so in the near-term. The need to establish future Canadian criteria and guidance will nevertheless be taken into consideration, pending further developments internationally in response to the Fukushima event. It is planned that relevant information will be incorporated in other emergency-specific procedural guidance being prepared by the CNSC, and not just exclusively in the context of environmental management systems guidance.

Part C: G-306, *Severe Accident Management Programs for Nuclear Reactors*

G-306 provides guidance for a licensee to develop, implement and maintain a severe accident management (SAM) program. G-306 was published in 2006 to address the need for planning for severe accidents, in accordance with generally accepted international practices.

This guidance addresses accident response provisions at Canadian plants for severe accidents. A SAM program provides an additional defence against the consequences of those accidents that fall beyond the scope of events considered in the reactor design basis. The establishment of a SAM program ensures that the personnel involved in managing an accident has the information, procedures, and resources necessary to carry out effective onsite actions.

The updated guidance in G-306 now includes:

- SAMGs that consider multiple-unit severe accident scenarios
- hydrogen mitigation
- equipment survivability
- adequate response to an extended station black-out and/or external events that have major impacts to the facility

Note: This amendment for G-306 is an interim measure, to provide guidance according to the lessons learned provided by the Fukushima Task Force, while a new regulatory document for accident management is developed.

Table C. G-306 Proposed Amendments and Rationale

G-306 Section #	Current Text	Proposed Changes	Rationale
Preface		**Preface** **This regulatory document sets out the expectations and guidance of the Canadian Nuclear Safety Commission (CNSC) with respect to severe accident management programs.** **When published, this document will amend/supercede G-306, *Severe Accident Management Programs for Nuclear Reactors*. This document has been amended to clarify or add criteria reflecting lessons learned from the Fukushima nuclear event of March 2011. The amendments were made to address findings from INFO-0824, *CNSC Fukushima Task Force Report*, as applicable to G-306.** --- This document may be used as part of the licensing basis for nuclear facilities and regulated activities, including when referenced in a licence, either directly or indirectly (through licensee reference documents). The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and thus establishes the basis for the CNSC's compliance program in respect of that regulated facility or activity. The licensing basis for a regulated facility or activity is a set of requirements and documents comprising:	To provide the administrative history of the amended document, the legal basis, and an explanation of the mandatory language in regulatory and guidance documents. The rationale for the amendment, as related to the *CNSC Fukushima Task Force Report,* is also provided.

G-306 Section #	Current Text	Proposed Changes	Rationale
		(i) the regulatory requirements set out in the applicable laws and regulations (ii) the conditions and the safety and control measures described in the facility's or activity's licence, along with 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 In this document, "shall" is used to express a requirement – i.e., a provision that a licensee or licence applicant is obliged to satisfy, in order to comply with the requirements of this regulatory document. "Should" is used to express guidance, or that which is advised. "May" is used to express an option, or that which is permissible within the limits of this regulatory document. "Can" is used to express possibility or capability. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee's responsibility to identify and comply with all applicable regulations and licence conditions.	
6.1	**Risk Assessment** The results of probabilistic risk assessment should assist the licensee to: 1. Verify that SAM would be effective for the severe accident sequences with the highest probability of occurrence, including natural and human-induced external hazards;	**Risk Assessment** The results of probabilistic risk assessment should assist the licensee to: 1. Verify that SAM would be effective for **representative** severe accident sequences, **including multi-unit events, events triggered by** natural and human-induced external hazards, **and extended station blackout accidents**;	Amends the text to address the following related to the FTF recommendations: To ensure that SAM is effective for multi-unit events and events triggered by external events. Considers events affecting multiple reactors on the site, events at spent fuel bays, as well as events triggered by extreme external hazards. Detailed assessments of the severe accident

G-306 Section #	Current Text	Proposed Changes	Rationale
			management procedural guidance and design capabilities include beyond-design-basis and severe accidents are a high priority. To demonstrate that revised emergency plans in regard to multi-unit accidents and severe external events, minimum complements, and emergency response organizations are capable and effective. It is currently demonstrated that emergency response organizations are capable of responding to single unit, beyond design basis events.
7.2	**Evaluation of Systems and Equipment** **If** systems and equipment are expected to perform in a way or under conditions that were not considered in their original design**,** **then** the licensee should conduct an assessment of their potential availability, effectiveness, and limitations for use in support of a SAM program. Existing systems may warrant design enhancement if the assessment reveals that the potential consequences of severe accidents are such that the existing systems may not provide the desired preventive and mitigating capabilities.	**Evaluation of Systems and Equipment** **Plant design capabilities for severe accident management – such as containment venting, hydrogen mitigation, and coolant make-up provisions – should be identified.** **For all** systems and equipment **which** are expected to perform in certain manners or conditions that were not considered in their original design, the licensee should conduct an assessment of their potential availability, effectiveness, and limitations for use in support of a SAM program. Existing systems may warrant design enhancement, if the assessment reveals that the potential consequences of severe accidents are such that the existing systems may not provide the desired preventive and mitigating capabilities. **Essential plant monitoring features and instrumentation for diagnosis of plant state should be identified, and verified to function reliably and provide meaningful data under**	Amends the text to address the following related to the FTF recommendations: To identify and evaluate the effectiveness and survivability of equipment needed to mitigate challenges on containment integrity and minimize consequences of a severe accident. To cover the installation of passive autocatalytic recombiners. To demonstrate key instrumentation is fully qualified for design-basis accidents, survivability and beyond-design-basis accident conditions as it is for DBA. To demonstrate that the minimum Class I/II equipment that is needed to mitigate beyond-design-basis accidents involving loss of all AC power is systematically identified. To ensure plant design capabilities for severe accident management, such as containment venting, hydrogen mitigation, coolant make-up provisions, instrumentation, and the control areas are evaluated and documented. Such design

G-306 Section #	Current Text	Proposed Changes	Rationale
		severe accident conditions.	capabilities would allow minimization of the consequences of a severe accident, should one occur. Demonstrates that requirements for design of systems credited in management of BDBAs are adequate, particularly for severe accident harsh environments (e.g., battery life, availability of portable instruments, connections to portable pumps for heat sinks, capability to re-energize instrumentation supplies). Demonstrated compliance to requirements for complementary design features that could be called upon to protect the containment, such as filtered containment venting.
7.3	**Assessment of Material Resources** The licensee should perform an assessment to determine the availability of coolant, energy, and other material resources that may be required for the effective completion of SAM actions.	**Assessment of Material Resources** The licensee should perform an assessment to determine the availability of coolant, energy, and other material resources that may be required for the effective completion of SAM actions. **For procurement of external resources (equipment, power, water and staff), the licensee should assess the adequacy of arrangements with other organizations, to ensure availability, timing and access to these resources during accidents, with consideration of potential challenges posed by common cause/external events. These arrangements should be formalized and documented.**	Amends the text to address the following related to the FTF recommendations: To require demonstrating adequacy of arrangements for procurement of external resources (equipment, power, water and staff) in terms of timing, access, availability. To demonstrate that licensees' emergency response organizations have access to a regional warehouse that could make available offsite equipment and resources that may be needed in case of a severe accident. Availability of emergency equipment could allow terminating a severe accident early enough to prevent any radioactive releases to the environment. To demonstrate that arrangements and agreements for external support formalized and documented in the applicable emergency plans and procedures.

G-306 Section #	Current Text	Proposed Changes	Rationale
9.2	**Personnel Training** The licensee should provide operating staff and emergency groups with training commensurate with their respective roles in accident management, enabling them to: 1. Understand their roles and responsibilities within the SAM program; 2. Learn about severe accident phenomena and processes; 3. Become familiar with the activities to be carried out; 4. Enhance their ability to perform in stressful conditions; and 5. Verify the effectiveness and improve the clarity of SAM procedures and guidelines. Training programs should address the roles to be performed by the different groups, and include drills and exercises to enable assessment of the interactions between the various groups involved in SAM. To the extent practicable, the licensee should use simulator training, because it provides a realistic and interactive environment and is an efficient method for enhancing human response in complex situations.	**Personnel Training** The licensee should provide operating staff and emergency groups with training commensurate with their respective roles in accident management, enabling them to: 1. understand their roles and responsibilities within the SAM program 2. learn about severe accident phenomena and processes 3. become familiar with the activities to be carried out 4. enhance their ability to perform in stressful conditions 5. verify the effectiveness and improve the clarity of SAM procedures and guidelines Training programs should address the roles to be performed by different groups, and include drills and exercises to enable assessment of the interactions between the various groups involved in SAM. **The licensee should develop a set of drills to cover multi-unit events and events triggered by external events.** To the extent practicable, the licensee should use simulator training, because it provides a realistic and interactive environment and is an efficient method for enhancing human response in complex situations.	Amends the text to address the following related to the FTF recommendations: To ensure that SAM is effective for multi-unit events and events triggered by external events. Considers events affecting multiple reactors on the site, events at spent fuel bays, as well as events triggered by extreme external hazards. Detailed assessments of the severe accident management procedural guidance and design capabilities include beyond design basis, and severe accidents are a high priority. To ensure plant design capabilities for severe accident management, such as containment venting, hydrogen mitigation, coolant make-up provisions, instrumentation, and the control areas are evaluated and documented. Such design capabilities would allow minimization of the consequences of a severe accident, should one occur. Ensures that requirements for design of systems credited in management of BDBAs are adequate, particularly for severe accident harsh environments (e.g., battery life, availability of portable instruments, connections to portable pumps for heat sinks, capability to re-energize instrumentation supplies). To demonstrate that revised emergency plans in regard to multi-unit accidents and severe external events, minimum complements, and emergency response organizations are capable and effective. It is currently demonstrated that emergency response organizations are capable of responding to single-unit beyond design basis events.

G-306 Section #	Current Text	Proposed Changes	Rationale
			To demonstrate that the performance of the emergency response organization under severe event and/or multi-unit accident conditions has not been challenged by designing and conducting exercises that are based on such conditions.
10.0	**Validation and review** The licensee should validate a SAM program, upon its establishment, to confirm its effectiveness, usability, technical accuracy, and scope. This validation should include modeling of selected accident scenarios with and without consideration of accident management actions, as well as drills and exercises. The licensee should also perform periodic reviews of a SAM program, provisions, guidelines, and procedures to reflect changes in plant design, operational modes, or organizational responsibilities. The reviews should address new information that has been derived from drills, exercises, training programs, safety analyses, experimental research or other sources.	**Validation and review** The licensee should validate a SAM program upon its establishment, to confirm its effectiveness, usability, technical accuracy and scope. This validation should include modeling of selected accident scenarios with and without consideration of accident management actions, as well as drills and exercises. **A validation assessment should be undertaken, to confirm that operator actions are possible, accounting for variables such as ease of access, possible radiation fields, presence of debris, fires or flooding, and staff complement.** The licensee should also perform periodic reviews of a SAM program, provisions, guidelines and procedures, to reflect changes in plant design, operational modes, or organizational responsibilities. The reviews should address new information that has been derived from drills, exercises, training programs, safety analyses, experimental research or other sources.	Amends the text to address the following related to the FTF recommendations: To ensure that SAM is effective for multi-unit events and events triggered by external events. Considers events affecting multiple reactors on the site, events at spent fuel bays, as well as events triggered by extreme external hazards. Detailed assessments of the severe accident management procedural guidance and design capabilities include beyond design basis, and severe accidents are a high priority. To demonstrate that revised emergency plans in regard to multi-unit accidents and severe external events, minimum complements, and emergency response organizations are capable and effective. It is currently demonstrated that emergency response organizations are capable of responding to single-unit beyond design basis events. To demonstrate that the performance of the emergency response organization under severe event and/or multi-unit accident conditions has not been challenged by designing and conducting exercises that are based on such conditions.

G-306 Section #	Current Text	Proposed Changes	Rationale
Glossary	**Glossary**	**Glossary** **alternate AC power** **An alternating current power source that is available to, and located at (or nearby) a reactor facility, and is characterized by the following:** **1. is connectable to but not normally connected to the offsite or onsite standby and emergency AC power systems** **2. has minimum potential for common mode failure with offsite power to the onsite standby and emergency AC power sources** **3. is available in a timely manner after the onset of station blackout** **4. has sufficient capacity and reliability for operating all the systems required for coping with station blackout, and for the duration of time required to bring and maintain the plant in a safe shutdown state.** **station blackout (SBO)** **A complete loss of alternating current (AC) power from offsite and onsite main generator, standby and emergency power sources. Note that it does not include failure of uninterruptible AC power supplies (UPS) and DC power supplies. It also does not include failure of alternate AC power.** **Note: See also definition for alternate AC power in this document.**	New or modified definitions are provided.

Part D: RD-308, *Deterministic Safety Analysis for Small Reactor Facilities*

Note: Although this document was not specifically discussed in the *CNSC Fukushima Task Force Report*, the scope of work for RD-308 is identical to that for RD-310.

RD-308 sets out requirements for the development, implementation and maintenance of safety analysis for a small reactor facility. RD-308 provides requirements for deterministic analysis, with an emphasis on design basis accidents.

The updated criteria in RD-308 include:

- analyses that include multi-unit accidents and events
- assessments of potential "cliff-edges[1]" and associated margins
- analyses specifically designed to determine capacities of make-up water or reserve electricity in the event of multiple system failures

The new requirements ensure that these types of accidents and events are included in the analysis, and are not screened as having a frequency too low to be considered. It now also includes determination of "cliff-edges" and margins, in addition to demonstrating that safety goals are met.

Requirements and guidance are added to provide additional criteria for design basis accident and for beyond design basis accidents.

[1] A cliff-edge effect is a large change in consequences caused by a small change of conditions. [See the newly updated definition below]

Table D. RD-308 Proposed Amendments and Rationale

RD-308 Section #	Current Text	Proposed Changes	Rationale
Preface	**Preface** This regulatory document sets out the requirements of the Canadian Nuclear Safety Commission (CNSC) with respect to deterministic safety analysis for small rector facilities, which must be submitted to the CNSC pursuant to the *General Nuclear Safety and Control Regulations* and *Class 1 Nuclear Facilities Regulations*. RD-308, *Deterministic Safety Analysis for Small Reactor Facilities* identifies regulatory criteria for the preparation and presentation of a deterministic safety analysis for a small reactor facility. A small reactor facility contains a reactor with a power level of less than approximately 200 megawatts thermal (MWt), used for research, isotope production, steam generation, electricity production or other applications. This document establishes a modern risk-informed approach to the classification of accidents, one that considers a full spectrum of possible events, including the events of greatest consequence to the public. The document allows the use of a graded approach to determine the scope and depth of deterministic safety analysis. The CNSC expects applicants for new small reactor facility licenses to apply this regulatory document. For currently licensed	**Preface** This regulatory document sets out the requirements of the Canadian Nuclear Safety Commission (CNSC) with respect to deterministic safety analysis, which must be submitted to the CNSC pursuant to the *General Nuclear Safety and Control Regulations* and *Class I Nuclear Facilities Regulations*. **When published, this document will amend/supercede RD-308. This document has been amended to clarify or add criteria reflecting lessons learned from the Fukushima nuclear event of March 2011. The amendments were made to address findings from INFO-0824, *CNSC Fukushima Task Force Report*, as applicable to RD-308.** RD-308, *Deterministic Safety Analysis for Small Reactor Facilities,* identifies regulatory criteria for the preparation and presentation of a deterministic safety analysis for a regulated facility. A small reactor facility contains a reactor with a power level of less than approximately 200 megawatts thermal (MWt), used for research, isotope production, steam generation, electricity production or other applications. This document establishes a modern risk-informed approach to the classification of accidents, one that considers a full spectrum	To provide the administrative history of the amended document, the legal basis, and an explanation of the mandatory language in regulatory and guidance documents. The rationale for the amendment, as related to the *CNSC Fukushima Task Force Report,* is also provided.

RD-308 Section #	Current Text	Proposed Changes	Rationale
	small reactor facilities, CNSC expects the licensees to phase in the application of this document, to meet requirements to the extent practicable. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee's responsibility to identify and comply with all applicable regulations and licence conditions.	of possible events, including the events of greatest consequence to the public. The document allows the use of a graded approach to determine the scope and depth of deterministic safety analysis. The CNSC expects applicants for new small reactor facility licenses to apply this regulatory document. For currently licensed small reactor facilities, CNSC expects the licensees to phase in the application of this document, in order to meet its requirements to the extent practicable. -- This document may be used as part of the licensing basis for nuclear facilities and regulated activities, including when referenced in a licence, either directly or indirectly (through licensee reference documents). The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and thus establishes the basis for the CNSC's compliance program in respect of that regulated facility or activity. 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 the safety and control measures described in the facility's or	

RD-308 Section #	Current Text	Proposed Changes	Rationale
		activity's licence, along with 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 In this document, “shall” is used to express a requirement – i.e., a provision that a licensee or licence applicant is obliged to satisfy, in order to comply with the requirements of this regulatory document. “Should” is used to express guidance, or that which is advised. “May” is used to express an option, or that which is permissible within the limits of this regulatory document. “Can” is used to express possibility or capability. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee’s responsibility to identify and comply with all applicable regulations and licence conditions.	
4.2.1	**Identifying events** The licensee or applicant shall use a systematic process to identify postulated initiating events (including criticality events), event sequences and event combinations (“events” hereafter in this document) that can potentially challenge the safety functions of the reactor facility. This process must consider regulatory requirements and guidance, past licensing precedents, operational experience,	**Identifying events** The licensee or applicant shall use a systematic process to identify postulated initiating events (including criticality events), event sequences and event combinations (“events” hereafter in this document) that can potentially challenge the safety functions of the reactor facility **The licensee shall also identify events that may potentially lead to fission product releases, including those related to irradiated fuel pools and fuel**	Changes were made to: 1) clarify that any events potentially leading to fission product releases, even occurring outside the reactor, should be identified in order to be considered for safety analysis 2) extend the scope of analysis to include considerations of events that can potentially affect multiple reactors or related facilities at a site.

RD-308 Section #	Current Text	Proposed Changes	Rationale
	engineering judgment, results of deterministic and PSA and systematic review of the design. The identification of events shall account for: • all operating configurations, such as start-up, at-power operation, shutdown, maintenance, testing, surveillance, and refuelling • configurations and uses of the reactor facility • interactions between the reactor and any experimental devices, including: a. administrative procedures b. controls c. additional equipment related to the experimental devices ...	**handling systems.** This process must consider regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and PSA and systematic review of the design. The identification of events shall account for: • all operating configurations, such as startup, at-power operation, shutdown, maintenance, testing, surveillance, and refuelling • configurations and uses of the reactor facility • interactions between the reactor and any experimental devices, including: a. administrative procedures b. controls c. additional equipment related to the experimental devices **Common-cause events affecting multiple reactor units on a site, or a reactor unit and related facilities nearby, shall be considered.**	
4.2.2	**Scope of events analyzed** The list of events to be developed for the deterministic safety analysis shall include: • failures or malfunctions of SSCs • operator errors	**Scope of events analyzed** The list of events to be developed for the deterministic safety analysis shall include: • failures or malfunctions of SSCs • operator errors	Ensures that the identification of common-cause events takes into consideration events that can potentially affect multiple reactors at a site.

RDU-308 Section #	Current Text		Rationale
	• common-cause failures initiated by internal and external events	• common-cause failures initiated by internal and external events**, including those affecting multiple reactor units on a site**.	
4.3 4.3.3	**Acceptance Criteria** **4.3.3 Beyond design basis accidents** Safety analysis for **BDBAs** shall demonstrate that: • the reactor facility as designed is capable of meeting the safety goals as established in RD-367 • the accident management program is capable of providing mitigation for **BDBAs**, to the extent practicable	**Acceptance Criteria** **4.3.3 Beyond design basis accidents (BDBA)** Safety analysis for BDBAs shall demonstrate that: • the reactor facility as designed is capable of meeting the safety goals as established in RD-367 • the accident management program is capable of providing mitigation for BDBAs, to the extent practicable**, taking into account the long-term availability of cooling water, material and power supplies.**	Ensures considerations of long term make-up water and power supplies in the demonstration of meeting safety analysis acceptance criteria.
4.4.1	**Deterministic safety analysis method** The deterministic safety analysis method shall include: • conducting the calculations, including sensitivity cases, to predict the event transient, starting from the initial steady state up to the pre-defined end state	**Deterministic safety analysis method** The deterministic safety analysis method shall include: • conducting the calculations, including **performing** sensitivity **analysis and identifying, where necessary, margins to cliff-edge effects** • **an event should be analyzed** from its initial steady state up to the pre-defined **stable state in the long-term**	Ensures that (1) an event is continuously analysed up to the cold, depressurized state, and (2) cliff-edge margins are identified.

RD-308 Section #	Current Text	Proposed Changes	Rationale
4.4.2	**Deterministic safety analysis assumptions** The deterministic safety analysis for AOO and DBA (conservative analysis for level 3 defence in depth) shall: • account for the possibility of equipment being taken out of service for maintenance	**Deterministic safety analysis assumptions** The deterministic safety analysis for AOO and DBA (conservative analysis for level 3 defence in depth) shall: • incorporate the key input modeling parameter uncertainties, the key input plant parameters measurement uncertainties, and the measurement uncertainties for the actuation of mitigating systems; the uncertainties shall be properly estimated, following best national and international practices • apply the single-failure criterion to all safety groups, and ensure that the safety groups are environmentally qualified • use minimum allowable performance (as established in the OLCs) for safety groups • account for consequential failures that may occur as a result of the initiating event • credit the actions of process and control systems only where the systems are passive and environmentally qualified for the accident conditions • credit process systems only if they are already running and are not affected by the event • include the actions of process and control systems when their actions may have a detrimental effect on the consequences of the analyzed accident	Emphasizes that safety analysis should account for the potential unavailability of equipment that may be needed to maintain long-term stable cooling of the reactor, following an accident.

RD-308 Section #	Current Text	Proposed Changes	Rationale
		• consider the effects of aging on SSCs • account for the possibility of equipment being taken out of service for maintenance • **account for the possibility of equipment being rendered inoperable during a prolonged period when it is required to maintain the reactor at a stable state in the long-term, following an accident** • credit operator actions only when there are: a. unambiguous indications of the need for such actions b. adequate procedures and operator training for such actions c. sufficient time to perform the credited actions d. environmental conditions that do not prohibit such actions	
Glossary	**Glossary**	**Glossary** **cliff-edge effect** **A large increase in the severity of consequences caused by a small change of conditions. Note: Cliff-edges can be caused by changes in the characteristics of the environment, the event or changes in the plant response.**	New or modified definitions are provided.

Part E: RD-310, *Safety Analysis for Nuclear Power Plants*

RD-310 sets out requirements for the development, implementation and maintenance of safety analysis for a NPP. RD-310 provides requirements for deterministic analysis, with an emphasis on design basis accidents. RD-310 was published in 2008. An accompanying guide, GD-310 is published. See the CNSC Web site.

The updated criteria in RD-310 now include:

- analyses that include multi-unit accidents and events
- assessments of potential "cliff-edges"[2] and associated margins
- analyses specifically designed to determine capacities of make-up water or reserve electricity in the event of multiple system failures

The new requirements ensure that these types of accidents and events are included in the analysis, and are not screened as having a frequency too low to be considered. It now also includes determination of "cliff-edges" and margins, in addition to demonstrating that safety goals are met.

Requirements and guidance are added to provide additional criteria for design basis accident and for beyond design basis accidents.

[2] A cliff-edge effect is a large change in consequences caused by a small change of conditions. [See the newly updated definition below]

Table E. RD-310 Proposed Amendments and Rationale

RD-310 Section #	Current Text	Proposed Changes	Rationale
Preface	**Preface** This regulatory document was developed pursuant to the requirements and obligations set forth in the General Nuclear Safety and Control Regulations and in the Class I Nuclear Facilities Regulations, where a safety analysis report demonstrating the safety of the nuclear facility must be submitted to the Canadian Nuclear Safety Commission (CNSC). This document identifies high-level regulatory information for a nuclear power plant licence applicant's preparation and presentation of a safety analysis. The information required adheres to high standards and is consistent with modern national and international practices addressing issues and elements that control and enhance nuclear safety. In particular, it establishes a more modern risk-informed approach to the categorization of accidents, one that considers a full spectrum of possible events including the events of greatest consequence to the public. The CNSC expects proponents and applicants for new reactor licences to immediately apply this regulatory document in new-build submissions. In the context of existing reactors, CNSC	**Preface** This regulatory document was developed pursuant to the requirements and obligations set forth in the *General Nuclear Safety and Control Regulations* and in the *Class I Nuclear Facilities Regulations*, where a safety analysis report demonstrating the safety of the nuclear facility must be submitted to the Canadian Nuclear Safety Commission (CNSC). **When published, this document will amend/supercede RD-310, *Safety Analysis for Nuclear Power Plants*. This document has been amended to clarify or add criteria reflecting lessons learned from the Fukushima nuclear event of March 2011. The amendments were made to address findings from INFO-0824, *CNSC Fukushima Task Force Report*, as applicable to RD-310.** This document identifies high-level regulatory information for a licence applicant's preparation and presentation of a safety analysis. The information required adheres to high standards and is consistent with modern national and international practices addressing issues and elements that control and enhance nuclear safety. In particular, it establishes a more modern risk-informed approach to the categorization of accidents, one that considers a full spectrum of possible events including	To provide the administrative history of the amended document, the legal basis, and an explanation of the mandatory language in regulatory and guidance documents. The rationale for the amendment, as related to the *CNSC Fukushima Task Force Report,* is also provided.

RD-310 Section #	Current Text	Proposed Changes	Rationale
	expects the licensees to apply this document, in a graduated manner, to all relevant programs in future submissions.	the events of greatest consequence to the public. The CNSC expects proponents and applicants for new facility licences to immediately apply this regulatory document in new-build submissions. In the context of existing facilities, CNSC expects the licensees to apply this document, in a graduated manner, to all relevant programs in future submissions. -- This document may be used as part of the licensing basis for nuclear facilities and regulated activities, including when referenced in a licence, either directly or indirectly (through licensee reference documents). The licensing basis sets the boundary conditions for acceptable performance at a regulated facility or activity, and thus establishes the basis for the CNSC's compliance program in respect of that regulated facility or activity. 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 the safety and control measures described in the facility's or activity's licence, along with the documents directly referenced in that licence (iii) the safety and control measures described	

RD-310 Section #	Current Text	Proposed Changes	Rationale
		in the licence application, and the documents needed to support that licence application In this document, “shall” is used to express a requirement – i.e., a provision that a licensee or licence applicant is obliged to satisfy, in order to comply with the requirements of this regulatory document. “Should” is used to express guidance, or that which is advised. “May” is used to express an option, or that which is permissible within the limits of this regulatory document. “Can” is used to express possibility or capability. Nothing contained in this document is to be construed as relieving any licensee from any other pertinent requirements. It is the licensee’s responsibility to identify and comply with all applicable regulations and licence conditions.	
5.2.1	**Identifying Events** The licensee shall use a systematic process to identify events, event sequences, and event combinations (“events” hereafter in this document) that can potentially challenge the safety or control functions of the NPP. This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic and probabilistic assessments, and any other systematic reviews of the design.	**Identifying Events** The licensee shall use a systematic process to identify events, event sequences, and event combinations (“events” hereafter in this document) that can potentially challenge the safety or control functions of the NPP. **The licensee shall also identify events that may potentially lead to fission product releases, including those related to irradiated fuel pools and fuel handling systems.** This process shall be based on regulatory requirements and guidance, past licensing precedents, operational experience, engineering judgment, results of deterministic	Changes are made to: 1) clarify that any events potentially leading to fission product releases, even occurring outside the reactor, should be identified in order to be considered for safety analysis 2) extend the scope of analysis to include considerations of events that can potentially affect multiple reactors in a multiple unit station.

RD-310 Section #	Current Text	Proposed Changes	Rationale
	The identification of events shall account for all operating modes, and the list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. ...	and probabilistic assessments, and any other systematic reviews of the design. The identification of events shall account for all operating modes, **including low power operation and shutdown modes. Common-cause events affecting multiple reactor units on a site shall be considered. The** list of identified events shall be reviewed for completeness during the design and analysis process and modified as necessary. **...**	
5.2.2	**5.2.2 Scope of Events** The list of events identified for the safety analysis shall include all credible: 1. Component and system failures or malfunctions; 2. Operator errors; and 3. Common-cause internally and externally initiated events.	**5.2.2 Scope of Events** The list of events identified for the safety analysis shall include all credible: 1. component and system failures or malfunctions 2. operator errors 3. common-cause internally and externally initiated events**, including those affecting multiple reactor units on a site**	Ensures that the identification of common-cause events takes into consideration events that can potentially affect multiple reactors at a site.
5.3.3	**Acceptance Criteria** **5.3.3 Beyond Design Basis Accidents** Analysis for BDBAs shall be performed as part of the safety assessment to demonstrate that: 1. The nuclear power plant as designed can meet the established safety goals; and 2. The accident management program and	**Acceptance Criteria** **5.3.3 Beyond Design Basis Accidents** Analysis for BDBAs shall be performed as part of the safety assessment to demonstrate that: 1. The nuclear power plant, as designed, can meet the established safety goals. 2. The accident management program and	Ensures considerations of long term make-up water and power supplies in the demonstration of meeting safety analysis acceptance criteria.

RD-310 Section #	Current Text	Proposed Changes	Rationale
	design provisions, put in place to handle the accident management needs, are effective.	design provisions, put in place to handle the accident management needs, are effective, **taking into account the long-term availability of cooling water, material and power supplies.**	
5.4.2	**Analysis Method** The analysis method shall include the following elements: .... 6. Conducting calculations, including sensitivity cases, to predict the event transient, starting from the initial steady state up to the pre-defined end-state;	**Analysis Method** The analysis method shall include the following elements: … ... 6. Conducting calculations, including performing sensitivity **analysis and identifying, where necessary, margins to cliff-edge effects.** **7. An event should be analyzed** from its initial steady state up to the pre-defined **stable state in the long-term;** …	Ensures that (1) an event is continuously analysed up to the cold, depressurized state, and (2) cliff-edge margins are identified. The changes are at high-level, in line with RD-310, which provides only high-level requirements. Further guidance on long-term analysis can be found in accompanying document GD-310, as follows: ***5.4.2.6 Conducting calculations*** *The duration of the transients considered in the analysis should be sufficient to determine the event consequences. Therefore, the calculations for plant transients are extended beyond the point where the NPP has been brought to shutdown and stable core cooling, as established by some identified means (i.e., to the point where a long-term, stable state has been reached and is expected to remain as long as required). The analysis should take into account the capacity and limitations of long-term make-up water and electrical power supplies.*
5.4.4	**Analysis Assumptions** Assumptions made to simplify the analysis, as well as assumptions	**Analysis Assumptions** Assumptions made to simplify the analysis, as well as assumptions concerning the operating	Emphasizes that safety analysis should account for the potential unavailability of equipment that may be needed to maintain long-term stable cooling of the reactor,

RD-310 Section #	Current Text	Proposed Changes	Rationale
	concerning the operating mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. Apply the single-failure criterion to all safety systems and their support systems; 2. Account for consequential failures that may occur as a result of the initiating event; 3. Credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident; 4. Account for the possibility of the equipment being taken out of service for maintenance; and 5. Credit operator actions only when there are a) unambiguous indications of the need for such actions, b) adequate procedures and sufficient time to perform the required actions, and c) environmental conditions that do not prohibit such actions.	mode of the nuclear power plant, the availability and performance of the systems, and operator actions, shall be identified and justified. The analysis of AOO and DBA shall: 1. apply the single-failure criterion to all safety systems and their support systems 2. account for consequential failures that may occur as a result of the initiating event 3. credit actions of systems only when the systems are qualified for the accident conditions, or when their actions could have a detrimental effect on the consequences of the analyzed accident 4. account for the possibility of the equipment being taken out of service for maintenance **5. account for the possibility of the equipment being rendered inoperable during a prolonged period required to maintain the plant in a stable, cold and depressurized state, following an accident** 6. credit operator actions only when there are a) unambiguous indications of the need for such actions b) adequate procedures and sufficient time to perform the required actions c) environmental conditions that do not prohibit such actions	following an accident.

RD-310 Section #	Current Text	Proposed Changes	Rationale
Glossary	**Glossary**	**Glossary** **cliff-edge effect** **A large increase in the severity of consequences caused by a small change of conditions. Note: Cliff-edges can be caused by changes in the characteristics of the environment, the event or changes in the plant response.**	New or modified definitions are provided.