REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants* / Conception d'installations dotées de réacteurs : centrales nucléaires
Comments received from additional consultation / Commentaires reçus lors de la consultation supplémentaire

Comments received:
• during additional consultation (August 22 to October 21, 2013): 93 comments from three (3) reviewers

Commentaires reçus :
• lors de la consultation supplémentaire (du 22 août au 21 octobre 2013) : 93 commentaires reçus de trois (3) examinateurs

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1.	General	Bruce Power OPG	**Major:** Title change does not reflect that the intent of the document is the design of **NEW** nuclear power plants. Suggested Change: revert to previous document title. “Design of New Nuclear Power Plants” Application to new nuclear power plants is noted in document Preface but not reflected in title. Existing nuclear power plants are not designed to this REGDOC. This could give the wrong impression of application to the general public.	Agree that the requirements of this REGDOC do not apply to existing reactor facilities. This document is intended to be used to assess new license applications for reactor facilities. However, it also recommended that licensees of existing facilities use this document in a review of modern codes and standards if such a review is required. For this reason, the title of the document will remain “Physical Design: Nuclear Power Plants.” To provide additional clarification regarding the application of the document, the preface text has been revised as follows: 1. REGDOC-2.5.2, *Design of Reactor Facilities: Nuclear Power Plants*, sets out requirements and guidance for **new licence applications for** the design of new water-cooled nuclear power plants (NPPs or plants).” 2. “For proposed new facilities: This document will be used to assess new license applications for reactor facilities. Guidance contained in this document exists to inform the applicant, to elaborate further on the requirements or to provide direction to licensees and applicants on how to meet requirements. It also provides more information about approaches used by CNSC staff to evaluate specific problems or data during the review of licence applications. Licensees

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				are expected to review and consider guidance; if these are not being followed, the licensees should explain how the alternate approach they have chosen still meets regulatory requirements. **For existing facilities[1]:** The requirements contained in this document do not apply unless they have been included, in whole or in part, in the licensing basis. 1 Existing facilities in this document are effectively those first licensed before 2014"
2.	General	Bruce Power OPG	**Major**: There has been a significant replacement of "should" to "shall" throughout the document (from the original RD-337) which imposes a heavy burden of proof on the Licensee. Although the intent of the clauses generally remains the same, the implication is that any deviation would be non-compliance to the overall intent of the clause. Suggested Change: Recommend that the CNSC review all changes from "should" to "shall" from the earlier versions of the document and clarify the intent of the change. Additional burden to the industry is high without justification. Specific clauses that are highly recommended for change are included in the comments below.	Comment noted. The original (2008) version of RD-337, *Design of New Nuclear Power Plants*, did not contain any shall or should statements. One of the primary reasons for reviewing RD-337 was to clarify which provisions were intended to be mandatory through the implementation of standard mandatory ("shall") and non-mandatory ("should") language. These clarifications were highlighted in the consultation versions of draft RD-337 version 2 (2012) and draft GD-337 (*Guidance for the Design of New Nuclear Power Plants*, 2012). It should be noted that there were no substantive comments on the use of "shall" and "should" in the first round public comments. RD-337 version 2 and GD-337 were merged to form REGDOC-2.5.2. Merging the requirements and guidance has not led to any significant change in the use of "shall" and "should. The CNSC has reviewed specific comments related to the use of "shall" and "should". However, since earlier extensive internal and external review has not identified any systemic issues, a further global review is not considered necessary.

[1] Existing facilities in this document are effectively those first licensed before 2014.

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3.	General	Bruce Power OPG	**Major:** There is a great deal of overlap with requirements contained in other REGDOCs, with inconsistent wording between the documents. Suggested Change: Review this REGDOC and revise any requirements contained in other REGDOCs to align the wording and requirements or simply refer to the other REGDOC. Example see specific comment on section 4.2.1. Having requirements for multiple documents adds confusion. All requirements in one area should be contained in one document.	The practice of the CNSC is to limit overlap between regulatory documents. The CNSC has reviewed each identified instance of overlapping requirements and will assess the feasibility of targeted revisions to address these overlaps.
4.	General	Bruce Power OPG	**Clarification:** The document references the old RD/GD and S-294 documents throughout. Suggest Change: change references to the New REGDOCs reference e.g. REGDOC 2.4.2	The practice of the CNSC is to reference currently published regulatory documents, but for clarity the qualification "or successor documents" is included as noted below: 1. Cross references to S-294, *Probabilistic Safety Assessment (PSA) for Nuclear Power Plants* have been revised to add "or successor documents" to address the currently active project to update S-294. 2. Cross references to RD-310, *Safety Analysis for Nuclear Power Plants* have been revised to add "or successor documents" to address the currently active project to update RD-310. 3. Cross references to GD-310, *Guidance on Safety Analysis for Nuclear Power Plants* have been revised to add "or successor documents" to address the currently active project to update GD-310.
5.	General	Candu Energy	**Major:** Some of the referenced regulatory documents have been superseded or will shortly be superseded (i.e. RD/GD-310, S-294, G-306, G-225, RD-353 etc.) The latest editions of the regulatory documents should be	Agree. See response for comment #4.

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			referenced.	
6.	General	Bruce Power OPG	**Clarification**: Reference to CSA standards should be limited to the generic series designation and not include the specific year. Suggested Change: For example replace CSA N286 -05 with "CSA N286 or equivalent".	Agree. The specific years have been removed from the references to CSA standards.
7.	General "complimentar y design features"	Bruce Power OPG	**Clarification:** Consistent with current IAEA practice and industry recommendations, the term should be replaced with "additional safety features". For example: Section 6.1, 7.1, 7.2, 7.3.4, 8.6.12, and the Glossary Section. Suggested Change: Replace term "Complementary Design Features" with "Additional Safety Features". It may also be prudent to provide additional guidance on designing for DEC and additional safety features.	Agree that complementary design features may also be referred to as "additional safety features." A note has been added to the definition of complementary design features to indicate this. However, the term "complementary design features" has been retained in the body of the document because it cannot be readily mistaken for general English usage, which can be a problem with the term "additional safety features.
8.	2.0	Bruce Power OPG	**Major:** All companies have the opportunity to propose the use of alternative approaches to demonstrate compliance to the requirements outlined, as noted in Section 11. Alternative approaches are not restricted to non-water-cooled reactors, but the passage noted implies that this is the case. Suggest changing the section as noted for clarity. Suggested Change: "It is recognized that specific technologies may use alternative approaches. If a ~~design other than a water-cooled reactor~~ is to be considered for licensing in Canada, the design is subject to the safety objectives, high-level safety concepts and safety management requirements associated with this regulatory document. However, the CNSC's review of such a design will be undertaken on a case-by-case basis. *This includes designs that are non-water-cooled*." This section, in conjunction with Section 11, needs to be	Agree that alternative approaches are not restricted to non-water-cooled reactors. The first sentence of the referred paragraph, "it is recognized that any specific technologies may use alternative approaches," includes water-cooled and non-water-cooled technologies and therefore captures this intention. The second sentence applies only to non-water-cooled technologies.

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			clear that alternative approaches to demonstrate compliance may be undertaken for any technology on a case-by-case basis. This will allow for possible efficiencies when considering designs that were not necessarily designed to meet the RD-337/REGDOC 2.5.2 but meet the overall intent of specific clauses.	
9.	4.0	Bruce Power OPG	**Clarification:** Plant states are fundamental to the requirements presented in this guide, but are not discussed until section 7. Suggested Change: Move 7.2 and 7.3 or a summary ahead of 4.1. Better would be a new 'section 4.0 Plant States' in very simple form such as presented in CSA N290.0 Clause 4.2 and 4.3 and reference section 7 of REGDOC-2.5.2 for further detail. Safety Objectives could become section 5.	Agree. The following text has been added to section 4 (ahead of section 4.1): "The safety objectives and concepts described in this section apply to an NPP during operation or during an accident. Four common four plant states are defined: normal operation, anticipated operational occurrences (AOO); design-basis accidents (DBA); and beyond-design-basis accidents (BDBA). This document also introduces the plant state "design extension conditions" (DECs) as a subset of BDBAs that are considered in the plant design."
10.	4.1.1	Bruce Power OPG	**Clarification:** Mitigation is provided for accidents considered in the design. Suggested Change: "Provisions shall be made for the mitigation of the radiological consequences of any accidents considered in the design."	Agree. Text revised as suggested.
11.	4.2.1	Bruce Power OPG	**Major:** Requirements for deterministic safety analysis should be contained in REGDOC 2.4.1. Suggested Change: Replace all except 1st sentence. Replace balance with: "All requirements for deterministic safety analysis are provided in (REGDOC 2.4.1)". Having requirements for deterministic safety analysis in 2 different documents adds confusion.	Agree conceptually. The CNSC's practice is to limit overlap between regulatory documents. However, it is appropriate to locate the dose acceptance criteria for new NPPs in REGDOC-2.5.2 as it is an important consideration in the design of a new plant. It should be noted that the dose acceptance criteria for existing NPPs are located in licensing documents.
12.	4.2.2	Bruce Power OPG	**Major:** Simple summation of the event sequence frequencies is not necessarily the correct method to	Agree conceptually. Text has been added to the guidance section of 4.2.2 as follows:

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			aggregate. Suggested Change: Replace “the sum of frequencies of all event sequences” with “the aggregate of frequencies of event sequences” throughout. The method for aggregating uncertainties is of extreme importance and needs to be established in a manner which is acceptable to the industry and the CNSC.	The summation of frequencies for all event sequences required in the safety goals applies to internal events. These are events that originate in the facility. The aggregation of internal event and other hazard risk metrics performed through simple addition to demonstrate that the risk metrics (core damage frequency, small release frequency and large release frequency) are not exceeded might not be appropriate. It is recognized that when the risk metrics for external events are conservatively estimated, their summation with the risk metrics for internal events can lead to misinterpretation. Should the aggregated total exceed the safety goals, conclusions should not be derived from the aggregated total until the scope of the conservative bias in the other hazards is investigated.
13.	4.2.4	Bruce Power OPG	“The design shall facilitate the clear transfer of control between procedures for operational states, accident conditions, severe accident management and onsite emergency response.” **Clarification:** This addition is unclear. It is unclear how design can facilitate the transfer between various procedures. It is assumed that the intent of this clause is that the transfer of control between procedures can be completed quickly efficiently given the design. Suggested Change: “The design shall ~~facilitate~~ consider the ~~clear~~ transfer of control between procedures for operational states, accident conditions, severe accident management and onsite emergency response.”	Comment noted. The Merriam-Webster dictionary defines facilitate as “to make (something) easier.” In this context, the meaning is clear.
14.	4.3.3	Bruce Power OPG	**Clarification:** Agree that ALARA is important and should be considered in the design, but the OLC’s are based on the	Comment noted. Adopting the proposed wording would imply a relaxation to the requirements of the

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			design and corresponding safety analysis. Consideration should be given to ALARA, when appropriate. Consideration for ALARA in the OLCs is a “should” not a “shall” as the text of this section suggests. Suggested Change: “5. Requirements for surveillance, maintenance, testing and inspection of the plant to ensure that SSCs function as intended in the design. ~~and comply with the~~ Consideration should be given to the requirement for optimization by keeping radiation exposures ALARA”	*Radiation Protection Regulation*, which is beyond the scope of this project. To clarify this, the text has been revised as follows: “5. requirements for surveillance, maintenance, testing and inspection of the plant to ensure that SSCs function as intended in the design and comply with the requirement for optimization by keeping radiation exposures ALARA, **as per the *Radiation Protection Regulations***”
15.	5.1	Bruce Power OPG	**Clarification:** Terms not defined consistent with other regulatory requirements. Suggested Change: Consider definitions for “prescribed, designated, and classified.”	Agree that terms should be consistent with other regulatory requirements. Text revised as follows: “3. established the requisite security provisions in accordance with the *Nuclear Security Regulations* and associated regulatory documents”
16.	6.2	Bruce Power OPG	**Clarification:** Need clarification on the accident states: AOO, DBA, DEC, etc. Which states require all or only some of these functions? Some accident sequences impair or disable one or more of the functions but the overall safety goals may still be met. Clarity is required. Suggested Change: The identified safety functions apply “in all operational states and accident states, except where the accident involves loss of that function.	Agree. Note that the intent of this section is that accidents or accident sequences will not impair or disable these fundamental safety functions, although safety analysis may be performed assuming one or more of such functions are not available. Text revised as follows: “The fundamental safety functions apply in all operational states, DBAs and DECs, except where the postulated accident involves a loss of that function.”
17.	6.2	Candu Energy	**Clarification:** The term “SSCs important to safety” is defined in the glossary. Suggest making reference to this term in Section 6.2.	Comment noted. The phrase “SSCs necessary to fulfill safety functions…” is used in this section because the emphasis is on the safety functions. The CNSC agrees that these SSCs may coincide with SSCs important to safety, however it is not necessary

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				to use the term here.
18.	7.1	Bruce Power OPG	**Major:** Additional safety features/Portable equipment – such as emergency mitigating equipment, and pumps - should not necessarily constitute SSC important to safety (SiS). Analysis supporting design basis accidents defines the systems important to safety based on deterministic and probabilistic means, often including an element of engineering judgement. These systems are defined as "important to safety" as they ensure the unit is capable of meeting the safety goals. The inclusion of complementary design features as SiS extends the meaning of the definition to include BDBA/DEC analysis which may not have the same level of conservatism built into the analysis. Separate regulatory guidance is already in place to define SiS. It is agreed that additional safety features require special attention and periodic maintenance to ensure that they are capable of supporting any BDBA/DEC events, but the level of rigour and system qualification required for SiS may not be appropriate. This qualification incurs significant capital cost and maintenance burden where it may not be appropriate. Additionally, the reporting requirements associated with systems important to safety may not be appropriate for additional safety features. 1) Suggested change: Remove "complementary design features" as a criteria for selection of systems important to safety. **"**SSCs important to safety shall include: 1.safety systems ~~2.complementary design features~~ 3.safety support systems 4.other SSCs whose failure may lead to safety concerns (e.g., process and control systems) " 2) Add additional guidance to state that these systems	1). Agree that clarification is required with regards to the design requirements and safety analysis requirements as pertaining to complementary design features. This is addressed in the guidance provided below. 2). For better clarity, additional guidance has been provided as follows: "Although the probability of SSCs being called upon during DECs is very low, the failure of safety functions for the mitigation of DECs may lead to consequences with high severity. SSCs that provide these safety functions should be assigned a safety category commensurate with the safety significance. For certain complementary design features (such as onsite portable equipment) with high redundancy and extremely low probability of being called upon, a low safety class may be appropriate." 3) To better describe the difference between identification of SSCs important to safety and classification of SSCs, the following text has been added to the guidance: "Firstly, SSCs are identified as important or not important to safety. By virtue of their roles, safety systems. complementary design features and safety support systems will be identified as important to safety. Additionally, other SSCs that can have a significant impact on nuclear safety will also be identified as important to safety. After the SSCs important to safety are identified, they are classified. The safety classification

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			should be maintained and tested for high performance, but are not required to carry the qualification of systems important to safety. Recognize the current IAEA terminology to use “additional safety features” instead of “complementary design features” “Although the probability of SSCs being called upon during DECs is very low, the failure of safety functions for the mitigation of DECs may lead to consequences with high severity. Therefore, these safety functions should *be managed appropriately* ~~*assigned a safety category*~~ ~~commensurate with the safety significance.~~ *In the case of additional safety features whose only purpose is to support DEC events, a safety classification other than “important to safety” may be appropriate provided that there is reasonable assurance that the features will function as required when called upon.”* Note, that portable equipment is not considered under systems important to safety for existing nuclear power plants. Classification of additional safety features as a system important to safety is a significant change to the overall definition. This change will have a significant impact on the burden to the Owner (in terms of cost, maintenance, reporting, etc.) for minimal benefit to overall safety. Beyond design basis provisions can be managed through separate means.	considers a number of factors as listed above. The safety classification enables appropriate design rules to be selected as described in section 7.5.”
19.	7.1	Bruce Power OPG	**Major:** This definition needs to be consistent with RD/GD-98. Recommended Change: Revise to ensure consistency between the documents. Having requirements for Systems Important to Safety in multiple documents causes confusion. All requirements for	Comment noted. See response to comment #35.

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			Systems Important to Safety should be contained in RD/GD-98.	
20.	7.1	Bruce Power OPG	**Clarification:** Add clarity to distinguish from classification of pressure boundaries for safety systems (and other SSCs) in CSA N285.0. Suggested Change: Title of this section is revised to 'safety classification of structures, systems and components'.	Agree. Text revised as suggested.
21.	7.2 Figure 1	Bruce Power OPG	**Major:** Figure 1 is inconsistent with what Canada recommended to the IAEA in SSR2/1. Suggested Change: Use the figure shown below. (At end of comment table). A consistent methodology needs to be adopted to the extent possible, including IAEA documents.	Agree conceptually. Figure 1 has been revised and is consistent with what Canada recommended to the IAEA for use in SSR 2/1. See figures at the end of this table.
22.	7.2 Guidance	Bruce Power OPG	**Clarification:** Requires alignment with terminology in IAEA documents. Design authority is responsible for providing reasonable assurance. Suggested Change: "The design principles for <u>additional safety features</u> to deal with DECs do not necessarily need to incorporate the same degree of conservatism as those applied to the design up to and including DBAs. However, <u>the design authority should provide</u> reasonable assurance that the additional safety features will function when called upon."	Agree. Text revised as suggested, except for the replacement of the term "complementary design features," which has been maintained. As per response to comment #7, complementary design features may also be referred to as "additional safety features" and a note has been added to the definition of complementary design features to indicate this. Revised text reads as follows: "The design principles for complementary design features to deal with DECs do not necessarily need to incorporate the same degree of conservatism as those applied to the design up to and including DBAs. However, the design authority should provide reasonable assurance that the complementary design features will function when called upon."
23.	7.3	Candu Energy	**Major**: The description of "plant states" provided in Section 7.3 is not consistent with the definition provided in the glossary.	Agreed. Additional clarification is provided in the guidance. The first paragraph in the guidance has been changed as follows:

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			Section 7.3 of REGDOC-2.5.2, “Plant States”, states that the plant states are grouped into the following four categories: NO, AOO, DBA and DEC. The definition of “Plant States” in the Glossary states: “…Note: For the purpose of this document a plant is said to be in one of the following states: normal operation, AOO, DBA or BDBA (severe accidents and DEC are subsets of BDBA state)”. Suggest revising the definition in the glossary as follows (to be consistent with Section 7.3): “Note: For the purpose of this document, a plant is said to be in one of the following states: normal operation, AOO, DBA or DEC (where DEC are a subset of BDBA states and could include severe accidents).”	Plant states considered in the design are divided into normal operation, AOOs, DBAs and DECs. The design requirements of SSCs should then be developed to ensure that the plant is capable of meeting applicable deterministic and probabilistic requirements for each plant state. Note that the plant states diagram in section 7.2 identifies BDBA as a plant state. However, only a subset of BDBAs are considered in the design. These are DECs.
24.	7.3.4 Guidance	Bruce Power OPG	**Major:** "The portable equipment credited for……". This paragraph implies that portable equipment will match the safety classification of the SSC it is installed on. Suggested Change: It is recommended to remove this paragraph. If retained suggest rewording to: “The portable equipment credited for DECs are considered part of additional safety features.” Designating portable equipment as per the SSC when it comes to importance to safety may restrict procurement of this mitigating equipment impacting availability and flexibility of use.	Comment noted. To maintain consistency with section 7.1 as revised by comment #18, and to remove duplication, only the first sentence of the paragraph has been retained.
25.	7.3.4 Guidance	Bruce Power OPG	“The design should identify the features that are designed for prevention or mitigation of DECs. These features are called complementary design features and should:” **Clarification:** This wording implies that an entirely	1) Agree. Text revised as suggested. 2) Agree that complementary design features may also be referred to as “additional safety features.” A note has been added to the definition of complementary design features to indicate this. See

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			separate set of systems is required for prevention/mitigation of DECs and does not consider that other SSCs may also be used, when appropriate. The original wording of this section in GD-337 was acceptable. 1) Suggested Change: revise to be consistent with the original text in GD-337. “The design should identify the features that are designed for use in, or that are capable of preventing or mitigating events considered in DECs. These features include complementary design features and other SSCs that may be credited for DECs. These features should:” 2) It is further suggested that complementary design features be replaced by “additional safety features”, consistent with current IAEA terminology.	comment #7.
26.	7.3.4 Guidance	Bruce Power OPG	**Clarification:** Editorial. Missing additional indent for 5 sub-bullets Suggested Change: • “The necessary high confidence in low likelihood should, wherever possible, be supported by means such as: o multiple layers of protection o application of the safety principles of independence, diversity, separation, redundancy o use of passive safety features o use of multiple independent controls	Agree. Text revised as suggested.
27.	7.3.4 Guidance	Bruce Power OPG	**Clarification:** "Portable onsite……" Statement not clear. Suggested change: “Portable onsite or offsite equipment may be one of the means for mitigation in support of the Severe Accident Management Guidelines.”	Agree. Text revised as suggested.
28.	7.3.4.1	Bruce Power OPG	**Major:** For some low probability severe accidents (such as aircraft strikes, accidents which significantly impair the containment system, or multiple coincident events leading to containment failure) uncontrolled releases of	Comment noted. This section is intended to apply to DECs only. To clarify the scope of application, the text has been revised as follows:

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			radioactivity cannot be precluded based on the nature of these events. Agree that for DBAs and DEC, containment system should be able to preclude uncontrolled releases, but this is not necessarily true for all severe accidents. Suggested Change: "Containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of offsite emergency procedures following the onset of core damage. Containment shall also prevent uncontrolled releases of radioactivity after this period, *to the extent practicable***"** As written, compliance to the clause cannot be achieved for all severe accidents.	1. Title of subsection changed to "severe accidents within DEC" 2. "For DECs with severe core damage, the containment shall maintain its role as a leak-tight barrier for a period that allows sufficient time for the implementation of offsite emergency procedures following the onset of core damage. Containment shall also prevent uncontrolled releases of radioactivity after this period.
29.	7.3.4.1 Guidance	Bruce Power OPG	**Major:** Issue one: This is inconsistent with the way that severe accident analysis is carried out for CANDUs where we distinguish between normal leakage, enhanced leakage and gross failure. Enhanced leakage can lead to large releases without actually failing containment. Issue two: Filtered venting alone may not be sufficient. It could be that the method of filtering selected must work in conjunction with other features to ensure containment integrity. Suggested Change: *"Containment leakage in a severe accident should remain below the design leakage rate limit (as defined in section 8.6.4) for sufficient time to allow implementation of emergency measures. Beyond this time, containment leakage that would lead to exceeding the small and large release safety goals should be precluded, to the extent*	Issue 1: Comment noted. The guidance relates to the interpretation of leak-tightness in the requirement "The containment shall maintain its role as a leak-tight barrier…". For the containment of a new NPP that has been designed with specific provisions for management of severe accidents, the design leak rate seems a reasonable target. Issue 2: Partially agree. There is no need to add the words "to the extent practicable" since this is guidance. Text revised as follows: "Containment leakage in a severe accident should remain below the design leakage rate limit (as defined in section 8.6.4) for sufficient time to allow implementation of emergency measures. Beyond this time, containment leakage that would lead to exceeding the small and large release safety goals should be precluded. This may be achieved by provision of adequate filtered containment venting along with other features."

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			practicable. This may be achieved by provision of adequate filtered containment venting along with other features. Control of containment also requires that high leakage is controlled even before gross failure. This is key to limiting releases and to the design of filtered containment venting.	
30.	7.3.4.1 Guidance	Candu Energy	**Major:** Since severe accidents include accident conditions that may not be considered in the design, it may not be practical to preclude gross leakage that would lead to exceeding the small and large release safety goals. Also, containment integrity may need filtered containment venting to work in conjunction with other features, such as, hydrogen control systems and means to transfer heat to the ultimate heat sink. Suggest revising the text as follows (revised text is in bold font): "Containment leakage in a severe accident should remain below the design leakage rate limit (as defined in section 8.6.4) for sufficient time to allow implementation of emergency measures. Beyond this time, containment leakage that would lead to exceeding the small and large release safety goals should be precluded, **to the extent practicable**. This may be achieved by provision of adequate filtered containment venting along with other features."	Comment noted. See responses to comments #28 and 29.
31.	7.3.4.1 "Severe accidents"	Bruce Power OPG	**Clarification:** Clarity required on scope of this section. Suggested Change: 7.3.4.1 Severe Accidents be Included In Design Extension Conditions	Agree. This section applies to DECs only. The title of this subsection has been revised as per comment #28.
32.	7.3.4.1	Bruce Power OPG	**Clarification:** It needs to be clarified that, in some cases, the connection points may involve temporary connections. Suggested Change: "The design shall include redundant connection points to provide for water and electrical power which may be	Comment noted. The connection points are part of the permanently installed equipment and are part of the design. This requirement is not for the equipment that will be attached to the connection point.

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			needed to support severe accident management actions. This may include the use of temporary (non-permanent) connections which involve the temporary reconfiguration of existing equipment not intended specifically for that purpose.”	
33.	7.4.2	Bruce Power OPG	**Major:** This section mixes the terms External Hazards and External Events. The terminology needs to be applied consistently. E.g., Turbine missiles and internal fires are external events but not external hazards. The terminology is not consistent with REGDOC 2.4.2. Suggested Change: Use a consistent terminology throughout the document and across the REGDOCs. There needs to be internal consistency within this document and across the REGDOCs. Otherwise it will only lead to confusion.	Agree. “External event” has been changed to “external hazard” throughout section 7.4.2. To ensure consistency throughout the document, this change has also been made in sections 6.5, 7.15.1 paragraph 3, and 7.15.1 guidance on containment structure, paragraph 1.
34.	7.5	Bruce Power OPG	**Clarification:** Original text of the list of design rules in GD-337 was formatted differently, with several sections of sub-bullets. Suggested Change: Revert to original formatting in GD-337. • identified codes and standards • conservative safety margins • reliability and availability o material selection o single failure criterion o redundancy o diversity o independence o fail-safe design • equipment qualification o environmental qualification o seismic qualification o electromagnetic interference (EMI)	Agree. Text revised as suggested.

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			• operational considerations o testability o inspectability o maintainability o aging • management system	
35.	7.6	Bruce Power OPG	**Major:** The requirements for Reliability are covered in RD/GD98. Revise to ensure consistency between the documents. Suggested Change: All requirements for Reliability should be contained in RD/GD-98. Having requirements for Reliability in multiple documents adds confusion.	Comment noted. The requirements for reliability serve different objectives. RD/GD-98 emphasizes reliability programs during the normal operation phase to ensure SSCs meet their defined design and performance criteria at acceptable levels. REGDOC-2.5.2 is applied at the design stage to ensure that SSCs are designed with appropriate reliability.
36.	7.6	Candu Energy	**Major:** It would be useful to further clarify “Not all SSCs important to safety identified in the design phase will necessarily be included in the reliability program” to specifically state that the complementary design features will not necessarily be included in the reliability program, since the engineering rules in industry standards are intended for design basis conditions. Suggest revising text as follows (revised text is in bold font): “The design for reliability is based on meeting applicable regulatory requirements and industry standards. The design should provide assurance that the requirements of CNSC RD/GD-98, Reliability Programs for Nuclear Power Plants, will be met during operation. Not all SSCs important to safety identified in the design phase, **in particular complementary design features**, will necessarily be included in the reliability program. ”	Comment noted. Whether the complementary design features will be included in the reliability program required by RD/GD-98, *Reliability Programs for Nuclear Power Plants* should follow the requirements and guidance in RD/GD-98. Specific complementary design features may or may not fall into the scope of the RD/GD-98 reliability program.
37.	7.6	Bruce Power OPG	**Major:** Availability of safety systems requires clarification. Suggested Change: “The special safety systems shall be	Comment noted. This is a technology-neutral document. “Special safety systems” is a CANDU specific terminology; therefore, the current wording

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			designed to ensure an unavailability of $\leq 10^{-3}$ a/a " Availability of the special safety systems takes into account the function of the safety support systems. Support system design does not require such a stringent requirement.	to ask that the probability of a safety system failure on demand lower than 10^{-3} is appropriate.
38.	7.6 and Glossary	Bruce Power OPG	**Clarification:** Clarification and consistent use of terms of reliability and availability. Suggested Change: Using definitions in clause 3 of CSA N290.0 to glossary.	Partially agree. The definition of "availability" from CSA N290 has been added to the glossary of the document. The definition reads as follows: **Availability** The fraction of time that a component or system is able to function. "Availability" can also mean the probability that a component or system will be able to function at any given time. The definition of "reliability" has been maintained to be consistent with RD/GD-98, *Reliability Programs for Nuclear Power Plants*. The CNSC will consider amending the definition during the next revision of RD/GD-98.
39.	7.6.3	Bruce Power OPG	**Clarification:** The first sentence is the requirement the second sentence is more of a clarification. In many cases there is no single 'safe end state' for an SSC. Suggested Change: Add "To the greatest extent practicable" to the first sentence, and remove from second sentence. "The principle of fail-safe design shall be applied to the design of SSCs important to safety, to the greatest extent practicable."	Partially agree. The first sentence has been revised as follows to match IAEA SSR 2/1: "The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety." The second sentence has been retained as it provides additional and useful information to supplement the first sentence.
40.	7.6.5 item 1	Bruce Power OPG	**Clarification:** There are cases where this may not be met. The important thing is that the requirements of 7.6.5.1 are met for instrumentation. Suggested Change: Add "except for instrumentation when requirements of 7.6.5.1 are met" to the end of 7.6.5 sub-bullet (1).	Comment noted. The requirements in section 7.6.5 are intended to be applied to a single system performing both process functions and safety functions. The requirements of section 7.6.5.1 are intended to be applied to sharing between systems. If a system cannot meet item 1 of section 7.6.5, then design change may need to be made to separate the

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				process functions and safety functions into two separate systems. Note that the two systems may share some equipment as long as the requirements of section 7.6.5 (including section 7.6.5.1) are met.
41.	7.6.5.2	Bruce Power OPG	**Major:** Sharing of safety systems can be justified in certain circumstances, allowing support from one unit to an adjacent unit. Suggested Change: Move the discussion on exceptional circumstances to the guidance section, changing shall to should, and adding: "Justification for sharing of systems should be noted." Important safety benefits can be obtained through sharing of systems. Also cost-effective safety benefits can be achieved through this.	Agree conceptually. The second paragraph of section 7.6.5.2 has been revised to read as follows: "In exceptional cases when SSCs are shared between two or more reactors, such sharing shall exclude safety systems and turbine generator buildings that contain high-pressure steam and feedwater systems, **unless this contributes to enhanced safety**." The above change is aligned with IAEA SSR 2/1.
42.	7.7	Bruce Power OPG	**Major:** The concept that no active degradation mechanism should exist in the piping system to be qualified for LBB (leak before break) raises concerns with P/T LBB methodology where P/Ts are part of a piping system with known impact from degradation mechanisms. REG DOC should consider requirement for LBB on some key systems/ components (e.g. fuel channel pressure tubes), regardless of whether or not active degradation exists. Suggested Change: Removal of requirement. Alternatively: If requirement is retained, need to define active degradation. Suggest definition be along lines of: "Active degradation is determined through evidence that degradation of a type or magnitude not considered within the design basis of the component." There are many systems that have 'active' degradation (e.g. have some minimal level of corrosion in many systems), or could be considered to have 'active' degradation (some	Text revised as follows: "No **uncontrolled** active degradation mechanism should exist in piping systems to be qualified for LBB." The Leak-before-Break (LBB) methodology is primarily intended to remove dynamic loads the need for protective barriers resulting from large diameter PHTS pipe whip/jet impingement from the design basis of the surrounding safety systems. However, the LBB methodology, which helps confirm margins between leaking crack and critical crack coupled with a robust leak detection system, can be extended to fuel channels and feeders as an instrument for their defence in depth strategy. The reference to "no uncontrolled active degradation mechanism" aligns with requirements of US NRC SRP 3.6.3, Rev 1, which suggests two mitigation

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			have considered all piping is susceptible to fatigue crack initiation and growth, hence could be considered 'active degradation). The design process takes into account the known or reasonably postulated degradation mechanisms and demonstrates the component design is adequate for the planned service life. Leak Before Break is not a requirement of all pressure boundary codes and standards. In many cases the materials suitable for use in a pressure boundary circuit, effectively result in a flaw tolerant material, but there may be no specific requirements for a leak detection system with a minimum sensitivity. It is not clear how this rule would apply to key system, such as fuel channel pressure tubes, where having Leak detection capability (and LBB) is highly desirable, but component is known to degrade in-service. Pressure tube are known to deform under irradiation and thermal creep, deuterium ingress occurs, material properties changes, etc.	methods are needed to address materials susceptible to an active degradation mechanism such as stress corrosion cracking.
43.	7.7. Guidance	Candu Energy	**Major:** *"No active degradation mechanism should exist in the piping system to be qualified for LBB."* The concept that no active degradation mechanism should exist in the piping system to be qualified for LBB (leak before break) raises concerns with respect to pressure tube LBB methodology where pressure tubes are part of a piping system with known impact from degradation mechanisms. This bullet point raises greater uncertainty on the applicability of the well-proven methodology to monitor the condition of the pressure tubes as they age during in-service operation. Suggest removing bullet.	See response to comment #42.
44.	7.8	Bruce Power OPG	**Clarification:** Alignment with current practices with consideration of DECs is required.	Comment noted. "Bounding" has been replaced with the term "postulated" to better reflect the intention of this provision.

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			Suggested Change: "bounding" to "estimated"	
45.	7.8 Guidance	Candu Energy	**Major:** "the bounding harsh environment of DECs within each timeframe " Since DECs rely on use of best estimate analysis methods and nominal operating conditions, the intent of "bounding harsh environment" may be misinterpreted. Suggest revising text as follows: "the estimated representative harsh environment of DECs within each timeframe "	Comment noted. See response to comment #44. In addition, this document does not preclude using estimated bounding harsh environment of DECs.
46.	7.9.1	Bruce Power OPG	**Clarification:** Ability to connect testing equipment e.g. portable equipment is an industry need. Suggested Change: "If testing equipment is part of the safety system and stays connected when not in use for testing, the safety class of such equipment should be the same as for the safety system <u>unless reliable buffering is in place or system performance is not negatively impacted "</u>	Comment noted. Portable equipment should not be permanently kept connected to the safety systems; otherwise they should be part of safety systems. Modify the sentence to match IAEA SSR 2/1: "Test provisions that are permanently connected to safety systems should be part of the safety systems and should be the same class as the safety systems <u>unless reliable buffering is in place or system performance is not negatively impacted</u>."
47.	7.9.3	Bruce Power OPG	**Clarification**: Primary heat transport system level should be included in the list. Suggest including "HTS (Heat Transport System) level (where appropriate)".	Agree. The fourth item in the guidance list has been revised as follows: "reactor vessel water level for a light water reactor (LWR) , or heat transport system water level and moderator level for a CANDU reactor"
48.	7.10 Item 2	Bruce Power OPG	**Major:** Item #2 imposes timing requirements on operating action and establishing support of offsite services. Off-site support in an emergency in terms of resources and equipment can be relied upon earlier than 72 hours into an event.	Comment noted. This is a design requirement to ask that with a robust design, the plant will be able to maintain fundamental safety functions without the need for offsite services and support for at least 72 hours. This is also an important requirement to incorporate the Fukushima lessons learned. The

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			Suggested Change: Recommend removing qualifiers on Item 2 The emergency support systems shall: 1. be independent of normal and backup systems Replace item b with: b. relying on limited off site resources and support for 72 hours (or required mission time if < 72 hours). Recognize that response to events would include call in provision of resources and equipment to support.	CNSC recongizes that in practice, the off-site service and support would be obtained as early as possible and would likely be much earlier than 72 hours. The design must demonstrate, however, that this support is not necessary before 72 hours.
49.	7.10 Guidance Paragraph 3	Bruce Power OPG	**Major:** Standard industry requirements and experience align with 15 minutes for control room actions and 30 minutes for field activities. Suggested Change: Recommend requirement of 15 minutes / 30 minutes consistent with industry current requirements. This could pose additional costs on provisions beyond those required for safety. May impact safety analysis if credits extended.	Comment noted. See response for comment #52.
50.	7.10	Candu Energy	Major: "4. be testable under design load conditions" It may not be practical to test all emergency support systems under design load conditions. Suggest revising text as follows (revised text is in bold font): "4. be testable under design load conditions, **where practicable"**	Agree. Text revised as suggested.
51.	7.10	Bruce Power OPG	Clarification: "4. be testable under design load conditions"	Agree. Text revised as suggested.

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			This may not be practical under all conditions. Suggested Change: Add “where practicable” to this bullet.	
52.	7.10 Guidance 3rd paragraph	Candu Energy	**Major:** Imposing a requirement for a minimum of 30 minutes for operator actions in the control room and 1 hour in the field does not provide information that would enable a designer to justify alternative operator action times, based on operating experience and industry standards, e.g., ANSI/ANS-58.8-1994 (R2001, R2008), "Time Response Design Criteria for Safety Related Operator Actions" Suggest revising text as follows (revised text is in bold font): “Pre-installed equipment can be credited for accident mitigation after 30 minutes where only control room actions are needed or after 1 hour if field actions are needed. These actions should be limited to operating valves, starting pumps, etc. Guidance is provided in section 8.10.4 for justification of such actions. **Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field station.** **Additional information may be found in ANSI/ANS-58.8-1994 (R2001, R2008), "Time Response Design Criteria for Safety Related Operator Actions.”**	Agree. Text in section 8.10.4 revised as follows: “Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field location.” ANSI/ANS-58.8-1994 (R2001, R2008), "Time Response Design Criteria for Safety Related Operator Actions” has been added to the list of additional information for this section. See also comment No. 82.
53.	7.13.1	Bruce Power OPG	**Clarification:** Not all SSC important to safety need Seismic Qualification. Suggested Change: “The design authority shall ensure that seismically credited SSCs important to safety are qualified accordingly. This shall apply to:”	Comment noted. For new nuclear power plants, all SSCs important to safety require seismic qualification. Even SSCs which are part of a nuclear power plant and which are not safety related (e.g., turbine building) should have seismic qualification, but with different acceptance criteria compared to safety related SSCs.

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54.	7.13.1	Candu Energy	**Major:** Not all SSCs important to safety require seismic qualification. Suggest revising the text as follows: "*All SSCs that are required to be seismically qualified shall meet the requirements of Canadian national or equivalent standards.*"	Comment noted. See response to comment #53.
55.	7.13.1	Bruce Power OPG	**Clarification:** "Assessment" is a more practical word than "demonstrated " for BDBE Suggested Change: "A beyond-design-basis earthquake (BDBE) shall be identified that meets the requirements for identification of DECs as described in section 7.3.4. SSCs credited to function during and after a BDBE shall be <u>assessed as</u> capable of performing their intended function under the expected conditions. Such <u>assessment</u> shall provide high confidence of low probability of failure under BDBE conditions for these SSCs.	Comment noted. "Demonstrate" is used as it is interpreted by the CNSC to be less restrictive; for example, BDBEs may be demonstrated through testing or assessment.
56.	7.13.1	Candu Energy	**Major:** The last paragraph states: "*… SSCs credited to function during and after a BDBE shall be demonstrated to be capable of performing their intended function under the expected conditions. Such demonstration shall provide high confidence of low probability of failure under BDBE conditions for these SSCs.*" "Demonstrated" can be interpreted to mean that seismic qualification by testing for BDBE conditions is required, which is not consistent with industry standards. "Assessment" is more appropriate as a requirement. Suggest revising text to: "… SSCs credited to function during and after a BDBE shall be assessed to be capable of performing their intended function under the expected conditions. Such assessment shall provide high confidence of low probability of failure	Comment noted. "Demonstration" does not mean "seismic qualification by testing." To clarify this, text has been added to the document as follows: "This demonstration need not be seismic qualification by testing."

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			under BDBE conditions for these SSCs.”	
57.	7.13.1	Bruce Power OPG	**Clarification:** CSA N289 requirements would indicate a 1E-4 earthquake as typical for the DBE for a modern NPP. Multiplying the hazard by an additional factor may result in a different requirement from that indicated in CSA standards. Suggested Change: Align with requirements in CSA standards for design of NPP.	Comment noted. The requirements in this section for DBEs align with CSA N289; the additional requirements apply only to BDBEs, which are beyond the scope of N289.
58.	7.13.1	Bruce Power OPG	**Clarification:** The requirement that all SSCs important to safety be DBE qualified and that seismic fragility be evaluated for all SSCs important to safety should be limited to those credited for response to a seismic event. Suggested Change: Add: “for those SSCs credited for response to a seismic event” in the associated requirements in 7.13.1.	Comment noted. Application of the requirement in question (section 13.1, paragraph 1) is limited to the items identified in the list below the paragraph. Seismic fragility levels shall be evaluated for SSCs important to safety by analysis or, where possible, by testing. For new nuclear power plants, all SSCs important to safety require seismic qualification. Even SSCs which are part of a nuclear power plant and which are not safety related (e.g., turbine building) should have seismic qualification, but with different acceptance criteria compared to safety related SSCs.
59.	7.13.1 Guidance	Bruce Power OPG	**Clarification**: This is very detailed instructions, perhaps better positioned as non-mandatory requirements. Would most of this discussion be appropriately presented in an appendix? Is this discussion intended to clarify the content of CSA N289 or is this intended to reflect alternate design methods? If these are alternate methods, are they applicable for all plant states discussed (AOO, DBA, DEC,…)? Suggested Change: Revise to a higher level and make reference to a standard where appropriate.	Comment noted. The majority of information provided in section 7.13.1 is guidance provided given the lack of readily available information on this subject. For ease of readability, the guidance has been maintained in the body of the document.
60.	7.14	Bruce Power	**Clarification:** In-service testing, maintenance, repair, inspection and monitoring ***The majority of this section is covered by RD/GD-210.***	Comment noted. Agree that the second paragraph is covered by RD/GD-210, *Maintenance Programs for Nuclear Power Plants* and some CSA standards. The intention of including this information in REGDOC-

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			Suggested Change: Delete the 2nd paragraph: - "These activities shall be performed to standards commensurate with the importance of the respective safety functions of the SSCs, with no significant reduction in system availability or undue exposure of the site personnel to radiation."	2.5.2 is to allow designers to proactively identify these activities at the design stage and to ensure that these activities can be conducted without significant reduction in system availability or undue exposure of the site personnel to radiation.
61.	7.17	Candu Energy OPG	**Clarification:** 1. Reference should be made to REGDOC-2.6.3 once it is issued. 2. If requirements for "safety analysis" remain in REGDOC-2.6.3, then it should also be referenced under Section 9, "Safety Analysis" in REGDOC-2.5.2.	1. Agree. See response to comment #4. 2. Agree. See response to comment #4.
62.	7.20	Candu Energy	**Major:** The requirement in Section 7.20 is similar but not the same as the requirement in Section 2.3.6 of draft REGDOC-2.10.1: *"13. determine and implement methods for communicating with onsite personnel and offsite authorities, including the implementation of at least two levels of backup communications systems …"* Consistency and clarity in the requirements between the various CNSC regulatory documents is necessary. Suggest revising the text as follows: REGDOC-2.5.2 "The design shall ensure that methods of communication with appropriate backup are available within the NPP and in the immediate vicinity, as well as to offsite agencies, in accordance with the emergency response plan." REGDOC-2.10.1 "13. determine and implement methods of communication with appropriate backup within the NPP and in the immediate vicinity, as well as to offsite agencies …"	Agree that REGDOC-2.5.2 and 2.10.1 should be consistent. REGDOC-2.10.1 will be revised to match REGDOC-2.5.2.
63.	7.22.1 Guidance	Bruce Power OPG	**Major:** Vital areas are confidential information and should not be discussed in this REGDOC.	Agree that specifics regarding vital areas constitute confidential information. However, the inclusion of the statement, "The vital areas include the reactor

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			Suggest Change: removing the discussion on vital areas or rewording to: "*The identification of vital areas involves the identification and location of SSCs that require protection, in order to prevent significant radioactive releases. The vital areas are listed and described by each licensee within a controlled classified document. The protection measures for these identified vital areas should be assessed.*" For security purposes, industry suggests removal of this information from this document	building and the spent fuel pool" is not security-related information in and of itself as it does nto specify the physical location of this equipment. Therefore, this statement has been maintained in the document.
64.	8.4	Bruce Power OPG	**Major:** The clause as written is not technology neutral. Suggested Change: Reword to read: "The means for shutting down the reactor shall consist of at least two different systems to provide diversity." Requirement should be technology neutral.	Comment noted. The text is already technology-neutral as it does not require that both systems be fast acting. The proposed text does not include the requirements for separation and independence, which are important considerations for all technologies.
65.	8.4 Guidance	Bruce Power OPG	**Major:** Clarity of definition of "redundancy". Suggested Change: Align with usage in IAEA SSR-2/1 section 6.8. Terminology should be technology neutral and consistent with international standards.	Agree. Definition of redundancy added to glossary (wording from IAEA Safety Glossary, 2007). Definition reads as follows: **"redundancy** Provision of alternative (identical or diverse) structures, systems and components, so that any one can perform the required function regardless of the state of operation or failure of any other."
66.	8.4.1	Bruce Power OPG	**Clarification:** As noted in comments on REGDOC 2.4.1, industry requests that an allowance be given for a single individual parameter where multiple redundancies exist (e.g., Neutron Overpower with multiple detectors throughout the core, being able to effectively respond to a slow loss of regulation event. Suggest wording similar to wording being proposed for CSA N290.1. Suggested Change: Add an exception:	Comment noted. REGDOC-2.5.2 makes it clear that detailed requirements and guidance for trip parameters are provided in RD-310, *Safety Analysis for Nuclear Power Plants* and GD-310¸ *Guidance for Safety Analysis for Nuclear Power Plants* (or successor documents).

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			"if this is not practicable, single trip parameter coverage is acceptable if additional redundancy is provided for that parameter. (e.g., Neutron Overpower Trip with multiple detectors, for responding to a slow loss of regulation event)".	
67.	8.6.1	Bruce Power OPG	**Major:** The changes made to this clause impose design considerations on reactor technologies. Complementary design features (portable equipment) may not be required given the robustness of the design. Analysis for DECs will define the need/function of any complementary design features required. This is an example of "should" being changed to "shall" without consideration of the intent of the clause. Suggested Change: "The containment shall be a safety system and *~~shall~~ may include additional safety ~~complementary design~~ features, as required. Both ~~of which~~ the containment system and the additional safety features* shall be subject to the respective design requirements provided in this regulatory document." Requirement should be technology neutral.	Partially agree. Text revised as follows: "The containment shall be a safety system and may include complementary design features, as required. Both the containment system and the complementary design features shall be subject to the respective design requirements provided in this regulatory document."
68.	8.6.7	Bruce Power OPG	**Clarification:** Need clarity on "open airlock doors" during outages. This may be permitted, provided it can be demonstrated that appropriate recall strategies are in place to respond to a limiting outage accident condition. The ability to demonstrate that open airlock doors is acceptable during limiting outage or refurbishment condition will allow for easier personnel and equipment movement and will reduce outage times. Suggested Change: Add in the Guidance section: "Open airlock doors may be permitted, provided that it can be demonstrated that appropriate recall strategies are in place to respond to a limiting outage accident condition."	Agreed, however the suggestion of "open airlock doors during outages" relates to operational requirements and procedures rather than the design requirements. Therefore, no change is necessary.
69.	8.6.7	Candu Energy	**Clarification:** Clarification is needed on "open airlock doors" during outages. This may be permitted, provided it	Comment noted. See response to comment #68.

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			can be demonstrated that appropriate recall strategies are in place to respond to a limiting outage accident condition. The ability to demonstrate that open airlock doors is acceptable during limiting outage or refurbishment condition will allow for easier personnel and equipment movement and will reduce outage times. Suggest adding the following statement in the Guidance section: "Open airlock doors may be permitted, provided that it can be demonstrated that appropriate recall strategies are in place to respond to a limiting outage accident condition."	
70.	8.7	Bruce Power OPG	**Major:** Requiring that "acceptable conditions" are maintained in SSCs for a severe accident may not be practical. Severe accidents considered are only those within design extension conditions. Suggested Change: Delete bullet 1 or changing "shall" to "should" in the sentence "The design shall extend… After "… severe accident" add the words " considered within design extension conditions" Requirements should be consistent with other regulatory documents	Agree conceptually. Text revised as follows: "1. acceptable conditions can be maintained in SSCs **needed for mitigation of severe accidents.**
71.	8.7	Candu Energy	**Major:** 1. Suggest adding the following statement in the Guidance section: "Open airlock doors may be permitted, provided that it can be demonstrated that appropriate recall strategies are in place to respond to a limiting outage accident condition." 2. Suggest revising the text as follows:	1. Comment noted. See response to comment #68. 2. Agree. See response to comment #70.

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			“The design shall extend the capability to transfer residual heat from the core to an ultimate heat sink so that, in the event of a severe accident considered as a design extension condition: ”	
72.	8.8	Bruce Power OPG	**Major:** Depending on the design of the EHRS and the reactor in general, this system may not be required to mitigate the consequences of a DEC. Analysis for DECs will define the need/function of any systems required. Suggested Change: Move this statement into the Guidance section and re-word as follows: “There ~~shall~~ should be reasonable confidence that the EHRS will function during DECs, if required “ The clause, as written, imposes design constraints. Analysis for DECs will define the need/function of any systems required.	Partially agree. The EHRS may not always be called upon during DECs, but there shall be reasonable confidence that the EHRS will function once it has been called upon during DECs. Text revised as follows: “There shall be reasonable confidence that the EHRS will function during DECs, **if required.”**
73.	8.8	Candu Energy	**Clarification:** The term “passive” is not defined in the glossary. It is suggested that definition for passive component as provided in the IAEA safety glossary be used or adapted: “Passive component - A component whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.” Or “Passive - A function that does not depend on an external input such as actuation, mechanical movement or supply of power.”	Agree. Definition of “passive” from IAEA has been added to glossary. The definition reads as follows: **“Passive component** A component whose functioning does not depend on an external input such as actuation, mechanical movement or supply of power.”
74.	8.9 “house load operation”	Bruce Power OPG	**Clarification:** To ensure alignment, it is suggested that these terms be defined in the text and be included in the glossary.	Agree. The term “house load operating mode” has been replaced with “house load operation” as these terms are intended to mean the same thing. The term “house load operation” has been added to the

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	“house load operating mode”		Suggest Change: Provide a definition for the terms.	glossary. Definition reads as follows: **“House Load Operation**: Operation of a nuclear power plant in isolation from the grid and supplying power only to its own auxiliary electric loads.”
75.	8.10.1	Candu Energy	**Major:** The requirements for communications between the MCR, the emergency support facilities (i.e., emergency support centre and technical support centre) and to offsite emergency response organizations is implied in REGDOC-2.10.1 through the use of the term “emergency response facilities” and the explanatory bullet “technical support and control centres” in the guidance under Section 2.3.6. Please ensure consistency between REGDOC-2.5.2 and REGDOC-2.10.1.	Agree. The following terms have been revised to align with REGDOC-2.5.2 and REGDOC-2.10.1: 1. “onsite emergency support centre” in REGDOC-2.5.2 has been changed to “Onsite emergency response facility” 2. “technical support and control centre” has been changed in REGDOC-2.10.1 to “technical support centre”
76.	8.10.2	Candu Energy	**Major:** By only referring to the “ESC”, an implicit assumption is made that the technical support centre is located with the MCR and would be rendered unavailable at the same time as the MCR. The possibility of the technical support centre being separate and independent from the MCR should be considered. Suggest revising the text as follows: “The SCR shall be provided with secure communication channels to the ESC and to offsite emergency response organizations. If the technical support centre is available, there shall be secure communications channels from the SCR.”	Agree conceptually. Text revised as follows: “The SCR shall be provided with secure communication channels to the **emergency response facility** and to offsite emergency response organizations.”
77.	8.10.3	Bruce Power OPG	**Clarification:** The onsite ESC may not facilitate such things as determination of recommended public protective actions, the coordination of emergency response activities with federal, provincial, and municipal agencies, etc. Requirement should be consistent with REGDOC 2.10.1.	Agree. In addition, a reference has been added to the “additional information” section in 8.10 to G-225, Emergency Planning at Class I Nuclear Facilities and Uranium Mines and Mills, or successor document.

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			Suggest change: Refer to REGDOC 2.10.1 for requirements within all of section 8.10 instead of repeating here. Change to indicate that the "onsite ESC (or the TSC) is used to support the following functions…"	
78.	8.10.3	Bruce Power OPG	"The ESC shall include a SPDS similar to those in the MCR and in the SCR" **Clarification:** Requirement should be consistent with REGDOC 2.10.1. Suggested Change: See industry comments submitted on REGDOC 2.10.1.	Agree that requirement should be consistent with REGDOC-2.10.1. Note that REGDOC-2.10.1 provides requirements for the "display [of] nuclear facility data and other information." These requirements are high level so they can be applied to all facilities covered by the scope of REGDOC-2.10.1 (Class I facilities and uranium mines and mills). REGDOC-2.5.2 provides more specific requirements for a safety parameter display system (SPDS) that apply specifically to reactor facilities. These specific requirements are consistent with the high level requirements for display of nuclear facility data.
79.	8.10.3	Candu Energy	**Major:** The set of requirements for the emergency support facilities, as defined in Section 8.10.3 of REGDOC-2.5.2 do not appear to be fully consistent with the set of requirements for the emergency response facilities, as defined in Section 2.3.6 of REGDOC-2.10.1. Make the requirements in REGDOC-2.5.1 Section 8.10.3 fully consistent with REGDOC-2.10.1 Section 2.3.6.	Agree that requirement should be consistent with REGDOC-2.10.1. The CNSC will ensure that the requirements in REGDOC-2.10.1 are consistent with those in REGDOC-2.5.2.
80.	8.10.3	Candu Energy	**Clarification:** The Emergency support facilities should contain some guidance with regards to multiple unit sites. Suggest providing guidance with respect to multiple units at a site, such as: "The challenges with accident management activities when responding to common-cause events in multiple units should be identified and the emergency support facilities should be shown to be adequate."	Agree conceptually. Text has been added to guidance as follows: "In the case of plants with multiple units at a site, the emergency support facilities should be demonstrated to be adequate to respond to common-cause events in multiple units." In addition, a reference has been added to the "additional information" section in 8.10 to G-225, Emergency Planning at Class I Nuclear Facilities and

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			Additionally, this section should make reference to REGDOC-2.10.1.	Uranium Mines and Mills , or successor document.
81.	8.10.3	Candu Energy	**Major:** The issue is that the terminology in REGDOC-2.5.2 is not fully consistent with the terminology in REGDOC-2.10.1, i.e., “emergency support facilities” in REGDOC-2.5.2 and “emergency support centre” versus “emergency response facilities” in REGDOC-2.10.1. The guidance for Section 2.3.6 of REGDOC-2.10.1 indicates that the emergency response facilities includes the “technical support and control centres”, but does not make reference to the emergency support centre. Revise text to be consistent in the references to the emergency support centre, main control room, secondary control area and technical support centre in both REGDOC-2.5.2 and REGDOC-2.10.1.	Agree. Terminology has been revised to align REGDOC-2.10.1 and REGDOC-2.5.2. See comment #75.
82.	8.10.4	Bruce Power OPG	**Major:** The Canadian Industry standard of 15 minutes for operator action in the control room and 30 minutes for operator action outside of the control room has been shown to be safe and sufficient. There is no basis or justification provided to propose changing this fundamental credit. Suggested Change: “3. following indication of the necessity for operator action inside the control rooms MCR, there is at least 15 minutes available before the operator action is required 4. following indication of the necessity for operator action outside the control rooms MCR, there is a minimum of 30 minutes available before the operator action is required” Existing stations will not be able to meeting these requirements, if required to do so via the modification or refurbishment process. Although the clause does have a provision of ”alternative times” this may not be adequate for existing stations to demonstrate compliance. The requirement may also not be justified for new plants, where response within the standard 15 minutes/30 minutes	Comment noted. The proposed operator action times of 30 minutes inside the control room and 60 minutes outside the control room are reflective of current international practices and consistent with IAEA SSR-2/1. More specifically, countries and organizations such as the UK, France and the Western European Nuclear Regulators Association (WENRA) have the requirement of 30 minutes for operation actions inside control rooms. This is also consistent with ANSI/ANS-58.8 adopted by the U.S. and South Korea. Note that based on section 8.10.4, a minimum of 30 minutes for operator actions in the control room and 1 hour is only applied to accident conditions, including DBAs and DECs. ANSI/ANS-58.8-1994 only allows shorter operator action times for plant conditions with frequency higher than 10^{-2} per reactor year (AOOs), which corresponds with the operational states specified in section 7.2. The requirement for DBAs is for 15 minutes for

	Section	Organization	Comment	CNSC Response
			is acceptable.	diagnosis plus 10 minutes for action. In this regard, REGDOC-2.5.2 is consistent with ANSI/ANS-58.8-1994. IAEA SSR-2/1 provides high level requirements that sufficiently long time be available between detection and action times, although it does not specify the numeric values. It should also be noted that it is the CNSC's understanding that new NPPs (e.g., EC-6, AP1000 and EPR) intend to meet the 30/60 minute requirements. This REGDOC allows for alternative times, stating "Where justified, alternative action times may be used. The alternative action times should make due allowance for the complexity of the action to be taken, and the time needed for activities such as diagnosing the event and accessing the field location."
83.	8.12.1	Candu Energy	**Clarification:** Bullet 1.a. states: *"maintaining an approved subcriticality margin by physical means or processes, preferably by the use of geometrically safe configurations, under both normal and credible abnormal conditions"*. The term "credible abnormal conditions" should be defined. GD-369 defines "credible abnormal conditions" as: "events or event sequences with a frequency of occurrence equal to or more than 10-6 per year."	Agree. Unnecessary detail removed as suggested. The requirements text points to RD-327 for the CNSC requirements concerning criticality.
84.	9.2 Guidance	Bruce Power OPG	**Clarification:** The guidance incorrectly states the Class 1 regulatory requirement for the final safety report. Suggested Change: Replace draft wording as follows:	Agree. Text revised as suggested.

	Section	Organization	Comment	CNSC Response
			"A final safety analysis report ~~*reflecting the "as-built" design*~~ *demonstrating the adequacy of the design is* required for an application for a licence to operate a Class 1 nuclear facility."	
85.	9.4	Candu Energy	**Clarification:** The purpose of DSA was originally provided in Section 9.4 (at a high level). This has been removed from REGDOC-2.5.2 and is covered in REGDOC-2.4.1 with different wording. It should be noted that high level objectives have been provided for PSA in Section 3 of REGDOC-2.4.2. The purpose of DSA was originally provided in Section 9.4 (at a high level). This has been removed from REGDOC-2.5.2 and is covered in REGDOC-2.4.1 with different wording. It should be noted that high level objectives have been provided for PSA in Section 3 of REGDOC-2.4.2.	Comment noted. The text was replaced with the reference to REGDOC-2.4.1, *Deterministic Safety Analysis,* in order to minimize any future inconsistencies between the two documents.
86.	9.5	Candu Energy	**Clarification:** The purpose of DSA was originally provided in Section 9.5 (at a high level). This has been removed from REGDOC-2.5.2, although it has been provided in Section 3 of REGDOC-2.4.2. Suggest revising Section 9.5 to capture the high level objectives for PSA from Section 3 of REGDOC-2.4.2: "The objectives of the probabilistic safety assessment are: to provide a systematic analysis, to give confidence that the design will comply with the fundamental safety objectives to demonstrate that a balanced design has been achieved to provide confidence that small change of conditions which may lead to a catastrophic increase in the severity of consequences (cliff-edge effects) will be prevented to provide assessments of the probabilities of occurrence for severe core damage states, and assessments of the risks of major radioactive releases to the environment	Comment noted. The text was replaced with the reference to REGDOC-2.4.2, *Probabilistic Safety Analysis*, in order to minimize any future inconsistencies between the two documents.

	Section	Organization	Comment	CNSC Response
			to provide site-specific assessments of the probabilities of occurrence, and the consequences of external hazards to identify plant vulnerabilities and systems for which design improvements or modifications to operational procedures could reduce the probabilities of severe accidents, or mitigate their consequences to assess the adequacy of emergency procedures to provide insights into the severe accident management program"	
87.	10.2	Bruce Power OPG	**Major:** The principles of pollution prevention and continuous improvement do not necessarily require the use of a BATEA methodology to select the most appropriate cooling water technology. Other methodologies (such as cost-benefit analysis) may also be used, although there are benefits and detriments to using any particular methodology. Overall societal benefits/costs can be considered in the evaluation. It is recommended that the Licensee be permitted to select the most appropriate methodology for their given application provided that the critical environmental and social factors are considered. It is also recommended that these techniques can be applied to selection of the condenser cooling technology to be used. Suggested Changes: Move to guidance section and reword as below. "Technological options for the design of cooling water systems shall should consider BATEA principles in order to minimize adverse environmental impact. Technological option selected for the design of cooling water systems should minimize the impact on the environment to the extent practicable, taking social and economic factors into consideration."	Agree conceptually. Note that BATEA is a pollution prevention performance standard rather than a methodology. Text revised as follows: 1. In the requirements section: "Pollution prevention principles shall be applied when considering the technological design options for cooling water systems in order to minimize adverse environmental impact." In the guidance section: "Pollution prevention principles should be conducted through an assessment of various technological options in order to identify the technology and techniques that are BATEA. The technological option selected for the design of cooling water systems should minimize the impact on the environment to the extent practicable given nuclear safety requirements. The economically achievable assessment of a technology option is not determined on the basis of a specific project but rather at the industry level. Technical feasibility of an option depends upon site-specific conditions taking into account environmental risk and socio-economic factors. The technology option of choice should be

	Section	Organization	Comment	CNSC Response
			Update Guidance section per wording below: "The selected condenser cooling technology should incorporate consider the latest in mitigation technology and techniques, and evaluate the options." The recent CCW Option Assessment for OPG DNNP has demonstrated that the selection of a cooling water system is highly dependent on site conditions. Allowing the Licensee to select the most appropriate methodology for their site is highly advised.	the one that best balances costs with environmental benefits resulting from application of a structured process of options analysis (e.g. cost-benefit analysis, multi criteria decision analysis). It should include an assessment on: • the age of equipment and facilities involved • how the option is designed, built, maintained, operated and decommissioned • the process employed • the engineering aspects of the application of various types of control techniques • process changes • technological advances or changes in scientific knowledge and understanding • cost of achieving the environmental benefits or reducing the environmental impacts • social-economic factors • time limits for installation of new and existing plants • other environmental impacts (including energy requirements) • other such factors as deemed appropriate by the regulator"
88.	11	Bruce Power OPG	**Major:** Text clarifying the use of alternative approaches was removed with this revision. It is recommended that the text be re-instated. It is unclear why this text was removed. Suggested Change: Re-instate the following text: *4. application of the expectations requirements in this document would result in undue hardship or other costs that significantly exceed those contemplated when the regulatory document was adopted* Inclusion of this provision has merit in assessing alternatives.	Comment noted. Regulatory document P-242, *Considering Cost-Benefit Information*, states that "when conducting a proceeding for purposes of a decision under the *Nuclear Safety and Control Act* that involves a licence or an order, the Commission or its designated officers will consider relevant information on costs or benefits that is submitted by a person who is participating in the process." This principle applies to the regulatory framework.

	Section	Organization	Comment	CNSC Response
89.	Abbreviations “DBT”	Bruce Power OPG	**Clarification:** Suggest that DBT be defined as it is used in several sections of the document (e.g. App A) (Design Basis Threat) Suggested Change: Add DBT to the Abbreviations List	Agree. Text revised as suggested.
90.	Glossary	Bruce Power OPG	**Major:** There is no definition of External Hazards and Internal Hazards in the list. Suggested Change: Include definitions for External Hazards and Internal Hazards consistent with definitions provided in REGDOC 2.4.2. There needs to be internal consistency within this document and across the REGDOCs. Otherwise it will only lead to confusion.	Agree. Text revised as suggested.
91.	Glossary	Bruce Power OPG	**Major:** The glossary needs to be comprehensive and complete and consistent with other regulatory documents. Suggested Change: It is recommended the CNSC provide a common glossary document. There needs to be internal consistency within this document and across the REGDOCs. Otherwise it will only lead to confusion.	Agree that glossary needs to be comprehensive and consistent with other regulatory documents. The glossaries between REGDOCs-2.4.1, *Deterministic Safety Analysis*, 2.4.2, *Probabilistic Safety Analysis* and 2.5.2 have been aligned. See response to comment #92 and Appendix A for list of revised definitions.
92.	Glossary	Candu Energy	**Major:** There should be consistent terminology between the Omnibus changes (i.e. REGDOC- 2.4.1, 2.5.2, 2.4.2...) Some definitions contain minor differences; some major (i.e. accident, PIEs, external events, shutdown state, SSCs). The definition of external event from the glossary in REGDOC-2.4.2 is as follows: “An event unconnected with the operation of a facility or with the conduct of an activity and that could have an effect on the safety of the facility or activity. External events include internal hazards and	Agree. The definitions of internal and external hazards have been added to the glossary. The following definitions have been aligned to ensure consistency between REGDOC-2.5.2 and the Fukushima omnibus documents: • accident • common-cause failure • confinement boundary • containment • design-basis accident

	Section	Organization	Comment	CNSC Response
			external hazards." Internal and external hazards are also defined in the glossary inREGDOC-2.4.2. The definition provided in REGDOC-2.5.2 does not explicitly discuss internal and external hazards. Furthermore, internal and external hazards are not defined in the glossary in REGDOC-2.5.2, but are discussed in Sections 7.4.1 and 7.4.2. Suggest aligning the definitions across omnibus REGDOCs.	• deterministic safety analysis • external event • event hazard • internal event • internal hazard • items important to safety • postulated initiating event • probabilistic safety assessment • shutdown state • structures, systems and components • uncertainty analysis For the definitions for the above list of terms, please refer to Appendix A.
93.	Glossary	Candu Energy	**Clarification:** The term "items important to safety" is used throughout the document (Sections 4.2.3, 7.3, 7.6.1, 9.2 and the glossary), however the definition of this term is not provided in REGDOC-2.5.2. It should further be noted that this term is also used in REGDOC-2.4.1. Suggest adding the definition of "items important to safety" to the glossaries in REGDOC-2.5.2 and REGDOC-2.4.1. It is suggested that the following definition from the IAEA safety glossary for 'items important to safety' be used: "An item that is part of a safety group and/or whose malfunction failure could lead to radiation exposure of the site personnel or members of the public."	Agree. The definition of the term "items important to safety" has been added to the document and reads as follows: "An item that is part of a safety group and/or whose malfunction failure could lead to radiation exposure."

Figure suggested by industry

<table>
<tr><th colspan="2">Operational States</th><th colspan="4">Accident Conditions</th></tr>
<tr><td rowspan="3">Normal Operation</td><td rowspan="3">Anticipated Operational Occurrences</td><td rowspan="3">Design Basis Accidents</td><td colspan="3">Beyond Design Basis Accidents</td></tr>
<tr><td colspan="2">Design Extension Conditions</td><td rowspan="2">Additional Beyond Design Basis Accidents (including additional sequences that may evolve into Severe Accidents)*</td></tr>
<tr><td>No Core Melt</td><td>Severe Accidents (Core Melt)</td></tr>
<tr><td colspan="3">Design Basis</td><td colspan="2">Considered in Design</td><td></td></tr>
<tr><td colspan="6">Reduced Frequency of Occurrence ---></td></tr>
</table>

Figure provided by CNSC and sent by Canada to IAEA in comments on revisions to SSR-2/1:

<table>
<tr><td colspan="2">Operational States</td><td colspan="3">Accident Conditions</td><td></td></tr>
<tr><td rowspan="3">Normal Operation</td><td rowspan="3">Anticipated Operational Occurrence</td><td rowspan="3">Design Basis Accident</td><td colspan="3">Beyond Design Basis Accidents</td></tr>
<tr><td colspan="2">Design Extension Conditions</td><td>Practically Eliminated Conditions</td></tr>
<tr><td></td><td colspan="2">Severe Accidents</td></tr>
<tr><td colspan="3">Design Basis</td><td colspan="2">Design Extension</td><td>Not considered in design</td></tr>
</table>

Reducing frequency of occurrence ➔

Figure in revised REGDOC-2.5.2

<table>
<tr><td colspan="2">Operational states</td><td colspan="4">Accident conditions</td></tr>
<tr><td rowspan="3">Normal operation</td><td rowspan="3">Anticipated operational occurrence</td><td rowspan="3">Design-basis accident</td><td colspan="3">Beyond-design-basis accidents ➔</td></tr>
<tr><td colspan="2">Design extension conditions</td><td>Practically eliminated conditions ➔</td></tr>
<tr><td>No severe fuel degradation</td><td colspan="2">Severe accidents ➔</td></tr>
<tr><td colspan="3">Design basis</td><td colspan="2">Design extension</td><td>Not considered as design extension ➔</td></tr>
</table>

Reducing frequency of occurrence ➔

Appendix A – REGDOC-2.4.1, 2.4.2 and 2.5.2 Disposition Tables
Common Definitions

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
accident	Any unintended event, including operating errors, equipment failures or other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety.	Any unintended event (including operating errors, equipment failures or other mishaps), whose consequences or potential consequences of are not negligible from the point of view of protection or safety. Note: For the purposes of this document, accidents include design-basis accidents and beyond-design-basis accidents. Accidents exclude anticipated operational occurrences, which have negligible consequences from the perspective of protection or safety.	Any unintended event (including operating errors, equipment failures or other mishaps) the consequences or potential consequences of which are not negligible from the point of view of protection or safety. Note: For the purposes of this document, accidents include design-basis accidents and beyond-design-basis accidents. Accidents exclude anticipated operational occurrences, which have negligible consequences from the perspective of protection or safety.
common cause	A cause for a concurrent failure of two or more structures, systems or components; for example, natural phenomena (earthquakes, tornadoes, floods, etc.), design deficiency, manufacturing flaws, operation and maintenance errors, and human-induced destructive events.	N/A	Remove from 2.4.1
common-cause event		An event that leads to common-cause failures.	Remove from 2.5.2
common-cause failure	A concurrent failure of two or more structures, systems or components due to a single specific event or cause, such as natural phenomena (earthquakes, tornadoes, floods, etc.), design deficiency, manufacturing flaws, operation and maintenance errors, and human-induced destructive events.	A concurrent failure of two or more structures, systems or components due to a single specific event or cause, such as natural phenomena (earthquakes, tornadoes, floods etc.), design deficiency, manufacturing flaws, operation and maintenance errors, human induced destructive events and others.	A concurrent failure of two or more structures, systems or components due to a single specific event or cause, such as natural phenomena (earthquakes, tornadoes, floods, etc.), design deficiency, manufacturing flaws, operation and maintenance errors, and human-induced destructive events.

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
confinement		A continuous boundary without openings or penetrations (such as windows) that prevents the transport of gases or particulates out of the enclosed space.	Remove from 2.5.2
confinement boundary	A continuous boundary without openings or penetrations and that prevents the release of radioactive materials out of the enclosed space		A continuous boundary without openings or penetrations and that prevents the release of radioactive materials out of the enclosed space.
containment	A method or physical structure designed to prevent the release of radioactive substances. This term is typically used in power reactors documentation	A confinement structure designed to maintain confinement at both high temperature and pressures, and for which isolation valving on penetrations is permitted.	A method or physical structure designed to prevent the release of radioactive substances.
design-basis	The range of conditions and events taken into account in the design of structures, systems and components of a nuclear power plant or a nuclear facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits for the planned operation of safety systems. The design basis includes the design description, design manuals, design drawings and the safety analysis report.	The range of conditions and events taken explicitly into account in the design of the facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems.	The range of conditions and events taken explicitly into account in the design of the facility, according to established criteria, such that the facility can withstand them without exceeding authorized limits by the planned operation of safety systems.
design-basis accident (DBA)	Accident conditions for which a nuclear power plant or a reactor facility is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within regulated limits.	Accident conditions for which a nuclear power plant is designed, according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.	Accident conditions for which ~~a nuclear power plant or~~ a reactor facility is designed according to established design criteria, and for which damage to the fuel and the release of radioactive material are kept within authorized ~~regulated~~ limits.

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
deterministic safety analysis	An analysis of a nuclear power plant's or a reactor facility's response to an event performed using predetermined rules and assumptions (e.g., those concerning the initial facility operational state, availability and performance of the facility systems and operator actions). Deterministic safety analysis can use conservative or best-estimate methods.	An analysis of nuclear power plant responses to an event, performed using predetermined rules and assumptions (e.g., those concerning the initial operational state, availability and performance of the systems and operator actions). Deterministic analysis can use either conservative or best-estimate methods.	An analysis of a reactor facility's response to an event performed using predetermined rules and assumptions (e.g., those concerning the initial facility operational state, availability and performance of the facility systems and operator actions). Deterministic safety analysis can use conservative or best-estimate methods.
external event	(REGDOC-2.4.2) An event unconnected with the operation of a facility or with the conduct of an activity and that could have an effect on the safety of the facility or activity. External events include internal hazards and external hazards (REGDOC-2.4.1) used, not defined		Events unconnected with the operation of a facility or the conduct of an activity that could have an effect on the safety of the facility or activity. Note: Typical examples of external events for nuclear facilities include earthquakes, tornadoes, tsunamis and aircraft crashes.
external hazard	(REGDOC-2.4.2) Hazards that originate from the sources located outside the site of the nuclear power plant. Examples of external hazards are seismic hazards, external fires (e.g., fires affecting the site and originating from nearby forest fires), external floods, high winds and wind induced missiles, offsite transportation accidents, releases of toxic substances from offsite storage facilities, and severe weather conditions. (REGDOC-2.4.1) not used or defined		Definition is deleted as external hazards are a subset of external events. The examples are moved into the body text of the document as guidance as follows Examples of external hazards are seismic hazards, external fires (e.g., fires affecting the site and originating from nearby forest fires), external floods, high winds, offsite transportation accidents, releases of toxic substances from offsite storage facilities, and severe weather conditions.

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
internal event	(REGDOC-2.4.2) Any event that proceeds from a human error or from a failure of a structure, system or component. (REGDOC-2.4.1) not used or defined	An event internal to the nuclear power plant that results from human error or failure in a structure, system or component.	Any event that proceeds from a human error or from a failure of a structure, system or component.
internal hazard	(REGDOC-2.4.2) Hazards that originate from the sources located on the site of the nuclear power plant (both inside and outside plant buildings). Examples of internal hazards are internal fires, internal floods, turbine missiles, onsite transportation accidents and releases of toxic substances from onsite storage facilities. (REGDOC-2.4.1) not used or defined		Hazards that originate from the sources located on the site of the reactor facility (both inside and outside plant buildings). The examples are moved into the body text of the document as guidance as follows. Examples of internal hazards are internal fires, internal floods, turbine missiles, onsite transportation accidents and releases of toxic substances from onsite storage facilities.
items important to safety	(REGDOC-2.4.1) An item that is part of a safety group and/or whose malfunction failure could lead to radiation exposure (REGDOC-2.4.2) not used or defined		An item that is part of a safety group and/or whose malfunction failure could lead to radiation exposure

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
postulated initiating event	(REGDOC-2.4.1) An event identified in the design as leading to either an anticipated operational occurrence or accident conditions. A postulated initiating event is not necessarily an accident itself; rather, it is the event that initiates a sequence that may lead to an anticipated operational occurrence, a design-basis accident or a beyond-design-basis accident, depending on the additional failures that occur. (REGDOC-2.4.2) not used or defined	An event identified in the design as capable of leading to an anticipated operational occurrence, or a design-basis accident, or a beyond-design-basis accident. This means that a postulated initiating event is not necessarily an accident itself; rather it is the event that initiates a sequence that may lead to an anticipated operational occurrence, a design-basis accident, or a beyond-design-basis accident, depending on the additional failures that may occur.	An event identified in the design as capable of leading to either an anticipated operational occurrence or accident conditions. Note: A postulated initiating event is not necessarily an accident itself; rather, it is the event that initiates a sequence that may lead to an anticipated operational occurrence, a design-basis accident or a beyond-design-basis accident, depending on the additional failures that occur.

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
probabilistic safety assessment	(REGDOC-2.4.2) For a nuclear power plant or nuclear fission reactor, a comprehensive and integrated assessment of the safety of the reactor facility. The safety assessment considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates that provide a consistent measure of the safety of the reactor facility, as follows: • A level 1 PSA identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and massive fuel failures • A level 2 PSA starts from the level 1 results, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment • A level 3 PSA starts from the level 2 results , analyzes the distribution of radionuclides in the environment and evaluates the resulting effect on public health		A comprehensive and integrated assessment of a reactor facility. The safety assessment considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates that provide a consistent measure of the safety of the reactor facility as follows: • A level 1 PSA identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and massive fuel failures • A level 2 PSA starts from the level 1 results, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment • A level 3 PSA starts from the level 2 results , analyzes the distribution of radionuclides in the environment and evaluates the resulting effect on public health Note: A PSA may also be referred to as a probabilistic risk assessment

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
probabilistic safety assessment	(REGDOC-2.4.1) A comprehensive and integrated assessment of a reactor facility. The safety assessment considers the probability, progression and consequences of equipment failures or transient conditions, to derive numerical estimates that provide a consistent measure of the safety of the reactor facility as follows: • A level 1 PSA identifies and quantifies the sequences of events that may lead to the loss of core structural integrity and massive fuel failures • A level 2 PSA starts from the level 1 results, analyzes the containment behaviour, evaluates the radionuclides released from the failed fuel, and quantifies the releases to the environment • A level 3 PSA starts from the level 2 results , analyzes the distribution of radionuclides in the environment and evaluates the resulting effect on public health		See above
shutdown state	(REGDOC-2.4.2) Shutdown A subcritical reactor state with a defined margin to prevent a return to criticality without external actions. (REGDOC-2.4.1) Shutdown state A subcritical reactor state with a defined margin to prevent a return to criticality without external actions.	A state characterized by subcriticality of the reactor. At shutdown, automatic actuation of safety systems may be blocked and support systems may remain in abnormal configurations.	shutdown state A subcritical reactor state with a defined margin to prevent a return to criticality without external actions NB: Also change in REGDOC-2.4.2

Term	REGDOC-2.4.1 (and REGDOC-2.4.2 where identified)	REGDOC-2.5.2	Proposed definition for both documents
structures, systems and components	(REGDOC-2.4.1) A general term encompassing all of the elements of a facility or activity that contribute to protection and safety. Structures are the passive elements: buildings, vessels, shielding, etc. A system comprises several components, assembled in such a way as to perform a specific (active) function. A component is a discrete element of a system. Examples are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves. (REGDOC-2.4.2) used but not defined	A general term encompassing all of the elements of a facility or activity that contribute to protection and safety. Structures are the passive elements: buildings, vessels, shielding, etc. A system comprises several components, assembled in such a way as to perform a specific (active) function. A component is a discrete element of a system. Examples are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves, etc.	A general term encompassing all of the elements of a facility or activity that contribute to protection and safety. Note: Structures are the passive elements: buildings, vessels, shielding, etc. A system comprises several components, assembled in such a way as to perform a specific (active) function. A component is a discrete element of a system. Examples are wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves, etc.
uncertainty analysis	(REGDOC-2.4.2) The process of identifying and characterizing the sources of uncertainty in the analysis, evaluating their impact on the probabilistic safety assessment results, and developing, to the extent practicable, a quantitative measure of this impact. (REGDOC-2.4.1) The process of identifying and characterizing the sources of uncertainty in the safety analysis, evaluating their impact on the analysis results, and developing – to the extent practicable – a quantitative measure of this impact.		The process of identifying and characterizing the sources of uncertainty in the safety analysis, evaluating their impact on the analysis results, and developing – to the extent practicable – a quantitative measure of this impact.