Candu

November 16, 2012

UNRESTRICTED | ILLIMITÉ

53A-CECC12-0041L

Canadian Nuclear Safety Commission,
P.O. Box 1046, Station 'B',
280 Slater Street,
Ottawa, Ontario K1P 5S9

To Whom It May Concern,

Subject: **Candu Energy Inc.'s Comments on GD-337 "Guidance for the Design of New Nuclear Power Plants"**

Please find attached Candu Energy Inc.'s comments on the draft guidance document GD-337, "Guidance for the Design of New Nuclear Power Plants".

Sincerely,

Original signed by

Albert Lee
Manager,
Project Physics, Licensing & Safety
(905) 823-9040, Ext. 36415

Cc: D. Yang, B. Pilkington, F. Yee, J. Ballyk, N. Anghelidis

Attachment:

A. Candu Energy Inc.'s Comments on GD-337, "Guidance for the Design of New Nuclear Power Plants"

2285 Speakman Drive
Mississauga, ON Canada L5K 1B1
Tel: 905 823 9040
www.candu.com

Attachment A

Candu Energy Inc.'s Comments on GD-337, "Guidance on the Design of New Nuclear Power Plants"

#				
1	General		If it is decided to combine RD-337 with GD-337, following the model of RD/GD-360 ("Long Term Operation Management for NPP", currently open for consultation), the combined RD/GD-337 must be clearly structured to differentiate between: 1. the requirements that may be used as part of the licensing basis for a regulated facility or activity by reference in a licence; and 2. the expectations and guidance on how to meet the requirements.	If it is decided to combine RD-337 with GD-337, it is suggested that: 1. the requirements be identified as "normative" to define the statements as mandatory; and 2. the "expectations and guidance" be identified as "informative" to define the statements as a means to meet the requirements.
2	General		It does not seem appropriate to have this guidance document out for public comment before the associated regulatory document has been finalized and approved by the Commission.	It is suggested that GD-337 be revised after RD-337 has been finalized and approved, and then issued again for public consultation.
3	General		The comments made on draft RD-337 version 2 should be taken into consideration for revisions to GD-337.	The comments provided during the public consultation phase of draft RD-337 version 2 should be considered for revision to GD-337.
4	General		The term "Design Extension Conditions" is used throughout the document; the use of the term "Beyond Design Basis Accidents" is preferred by industry. The accepted terminology in use within the Canadian nuclear industry is "beyond design basis accidents". It is preferred that the IAEA term "design extension conditions" not be used.	If the term "design extension conditions" is adopted for new NPPs, GD-337 should provide explanations for the relationship between "design extension conditions" and "beyond design basis accidents". The CNSC should provide guidance on the principles and guidelines for applying engineering design rules to SSCs that are included in the nuclear power plant design to provide safety functions for "design

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			If the CNSC adopts the term "design extension conditions", it is suggested that the IAEA definition and use of "design extension conditions" from IAEA SSR-2/1 be adopted in its entirety. Additionally, consistent terminology for DEC should be used in RD-337. In particular, consistency between Sections 4.2.3, 7.3 and the definitions provided in the glossary are needed.	extension conditions". The CNSC should also provide guidance on the principles and guidelines for performing deterministic safety analyses for "design extension conditions".
5	General		The "Additional Information" sections in the document are very helpful as they identify standards acceptable to the CNSC for ensuring compliance.	It is recommended that the practice of including "Additional Information" sections be carried forward for other GDs & RD/GDs.
6	General		Many standards are referenced throughout the document, with the applicable edition dates. This is not recommended practice, because newer editions of the standards may be issued between revisions to GD-337.	It is suggested that the applicable edition dates not be included, or a statement be included regarding the use of the most recent editions of the standards.
7	Preface and Section 2	"...SSR 2/1, Safety of Nuclear Power Plants: Design…"	Editorial: The correct title of SSR-2/1 is "Specific Safety Requirements: Safety of Nuclear Power Plants: Design"	It is suggested that the title of the document be corrected to: "… SSR-2/1, **Specific Safety Requirements:** Safety of Nuclear Power Plants: Design..."
8	3	Bullet 5	The list of paragraphs from Section 5 and Section 6 of the Class I Nuclear Facilities Regulations appears to be incomplete. This version of GD-337 includes guidance that is applicable to paragraphs 5(k), 6(j) and 6(k), however these are not listed.	It is suggested that the final version of GD-337 be reviewed against the Class I Nuclear Facilities Regulations for completeness.
9	4.3.3		The text in Section 4.3.3 of GD-337 does not provide any guidance on the definitions of "safety limits" and "limiting settings for	It is suggested that an explanation of the terminologies used for OLCs be provided, in particular the terms "safety limits" and

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			safety systems", terms which are used in Section 4.3.3 of draft RD-337 version 2. By introducing the text on OLCs from IAEA Safety Guide NS-G-2.2 in Section 4.3.3 of draft RD-337 version 2, it is also necessary to include an explanation of the terminology of OLCs from NS-G-2.2.	"limiting settings for safety systems".
10	5.3	"Design control measures, in the form of processes, procedures and practices, include: • design initiation, specification of scope and planning • specification of design requirements • selection of suitably qualified and experienced staff • work control and planning of design activities • specification and control of design inputs • review of design concepts and selection • selection of design tools and computer software • conducting conceptual analysis • conducting detailed design and production	The bullets do not follow a "chronological" order. The design control measures listed here should follow the order in which the design activities progress from initiation to being ready for implementation, as described in CSA N286-05 (it should be noted that CSA N286 June 2012 has been issued and may supersede CSA N286-05). Some activities are addressed in multiple bullets. For example, planning of design activities is mentioned in both the 1st and 4th bullets. The activity described in the bullet "management of the design and control of design changes" is also addressed in the bullet "configuration management". In the bullet "conducting conceptual analysis", the type of analysis should be specified (i.e. safety, stress??). CSA N286 clearly indicates a conceptual safety analysis should be performed to assess the preferred design concept. The bullet "selection of suitably qualified and	Suggest revising the text as follows: "Design control measures, in the form of processes, procedures and practices, include: • design initiation, including identification of scope • work control and planning of design activities • selection of competent staff • identification and control of design inputs • establishing design requirements • evaluation of design concepts and selection of preferred concept • selection of design tools and computer software • conducting conceptual safety analysis to assess preferred design concept • conducting detailed design and production of design documentation and records • conducting detailed safety analysis to prove adequacy of detailed design • defining any limiting conditions for

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		of design documentation and records • conducting detailed safety analysis • defining any limiting conditions for safe operation • carrying out design verification and validation • independence of individuals or groups performing verifications, validations and approvals • configuration management • management of the design and control of design changes • identification and control of design interfaces”	experienced staff” may suggest that only experienced staff can perform design activities, whereas the CSA N286-05 requirement is for competent personnel to perform the design work assigned to them (competence includes, in addition to experience, education, training, skills and ability). It is suggested that all bullets in this section follow the same order as in CSA N286-05.	safe operation • carrying out design verification and validation • configuration management • identification and control of design interfaces”
11	5.3		Draft RD-337 version 2 states “**The computer software used for design and analysis calculations shall be qualified in accordance with applicable standards**.” By using the term “qualified in accordance with applicable standards” some confusion may be introduced, because the nuclear industry is more familiar with the use of verified and validated software, as defined in CSA N286.7.	Suggest adding the following text to Section 5.3: “The computer software used for design and analysis calculations shall be qualified in accordance with applicable standards. This shall be achieved by following industry standards for software, such as CSA N286.7, where qualified software: (a) is shown to be capable of addressing

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			For clarification it is suggested that the definition of "qualified software" from CSA N286.7.1-09 be included in GD-337 to provide clarification and guidance on the intent of "shall be qualified in accordance with applicable standards".	intended problems; (b) is adequately specified, which includes (i) documentation of requirements, design, characteristics, and limitations of use; and (ii) identification of all required tool components and their required attributes; (c) possesses attributes that have been demonstrated to satisfy all requirements; and (d) includes configuration management and change control."
12	6.1.1	"For independent effectiveness of the different levels of defence, any design features that aim at preventing an accident should not belong to the same level of defence as the design features that aim at mitigating the consequences of the accident."	This paragraph would be more appropriate at the end of Section 6.1, rather than at the end of Section 6.1.1. Section 6.1.1 discusses the physical barriers, whereas this paragraph is applicable to the design features for all levels of defence-in-depth.	It is suggested that this paragraph be moved to the end of Section 6.1.
13	6.5	"Generally, a larger exclusion zone would require more emergency response time and capability."	A larger exclusion zone should allow for somewhat more relaxed response time, since the public is further from the source of the radiological hazard. A larger exclusion zone may not require more emergency response capability.	Suggest that the text be revised as follows: "Generally, a larger exclusion zone would allow for more emergency response time."
14	6.5	**Evacuation needs**	Environmental factors also affect evacuation times (i.e. precipitation = slower evacuation). Environmental factors are not specifically addressed in this section, although they are taken into consideration in the nuclear emergency response plans.	Suggest that the following text be added: "Environmental factors which can affect the response times should be taken into consideration."

#	Section	Excerpt of Section	Industry Issue	Suggested Change
15	6.6.1	"As stated in draft RD-337 version 2, "*the design shall take due account of challenges to a multi-unit site.*"	The use of the term "multi-unit site" can lead to confusion. One can have a site with multiple units as part of a single build project, or the addition of one or more units to an existing site where one or more units are already in operation.	It is suggested that the term "multi-unit site" be replaced with "multiple units at a site" throughout this document.
16	7.1	"The method for classifying the safety significance of SSCs important to safety should be based primarily on deterministic methodologies, complemented (where appropriate) by probabilistic methods."	The use of engineering judgement in the safety classification process should be acknowledged.	Suggest revising the text as follows: "The method for classifying the safety significance of SSCs important to safety should be based primarily on deterministic methodologies, complemented (where appropriate) by probabilistic methods **and engineering judgement**."
17	7.1	"The SSC classification process should include the following activities:…… • identification of engineering design rules for classified SSCs……."	The SSC classification process should not include the identification of engineering design rules for classified SSCs. Once a safety class has been assigned to an SSC, the appropriate engineering design rules should be applied to the SSC. The basic concept should be that the SSC is designed such that: • the most frequent occurrences yield little or no adverse consequences to the public, and • the improbable extreme situation, having the potential for the greatest consequences to the public, have a low probability of occurrence.	Suggest revising the text by replacing the bullet "identification of engineering design rules for classified SSCs" with the following paragraph: "Once the safety classification of SSCs is established, corresponding engineering design rules should be specified and applied. These engineering design rules should ensure that the SSCs possess all the design features necessary to achieve the required ability to perform their designated safety function with a sufficiently low failure rate consistent with the safety analysis. The SSCs should be designed with sufficient robustness to ensure that no operational loads caused by postulated initiating events will adversely affect the ability of the SSCs to perform their designated safety functions."

#	Section	Excerpt of Section	Industry Issue	Suggested Change
18	7.1	"Some specific SSCs classification guidelines are given below:.... • as a general rule, supporting SSCs should be assigned to the same class as that of the frontline SSCs to be supported......."	This statement does not appropriately account for whether the failure of the supporting SSC has the same consequence on the frontline SSC as a failure of the frontline SSC.	Suggest deleting the text.
19	7.1	"Some specific SSCs classification guidelines are given below:... • if a particular SSC contributes to the performance of several safety functions of different categories, it should be assigned to the class corresponding to the highest safety category, requiring the most conservative design rules..."	The selection of engineering design rules for a SSC should be commensurate with the principles of achieving the required level of: • ability to perform its designated safety function with a sufficiently low failure rate consistent with the safety analysis, and • robustness to ensure that no operational loads caused by postulated initiating events will adversely affect the ability of the SSCs to perform their designated safety functions. This does not necessarily mean requiring the most conservative design rules.	Suggest revising the text as follows: "Some specific SSCs classification guidelines are given below:... • if a particular SSC contributes to the performance of several safety functions of different categories, it should be assigned to the class corresponding to the highest safety category, requiring the **commensurate** design rules..."
20	7.1	"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 high severity consequences. Therefore, these safety functions should be considered a high safety category."	The phrase "these safety functions should be considered a high safety category" needs clarification. The term "high safety category" is not well defined and different readers can arrive at different conclusions. In terms of safety significance, safety functions required to mitigate the consequences of design extension conditions	Suggest revising the text 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 high severity consequences. Therefore, these safety functions should be **assigned a safety category commensurate with the safety**

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			should be ranked lower than: In terms of safety significance, safety functions required to mitigate the consequences of design extension conditions should be ranked lower than: • safety functions required to be performed immediately to control or mitigate the consequences of anticipated operational occurrences or design basis accidents; and • safety functions required to reach and maintain a stable safe shutdown condition.	**significance.**"
21	7.1		Draft RD-337 version 2 states that complementary design features are included in the list of systems important to safety. Portable equipment – such as emergency mitigating equipment, and pumps should not necessarily constitute systems important to safety. More clarification is required on positioning portable equipment under systems important to safety in complementary design features for new nuclear power plants. Note, that portable equipment is not considered under systems important to safety for existing nuclear power plants.	Suggest providing clarification on positioning portable equipment under systems important to safety in complementary design features for new nuclear power plants.
22	7.2		Draft RD-337 version 2 section 7.4.1 shows internal events can be classified as AOO, DBA or DEC; and RD-337 version 2 section 7.4.2 shows external events can be classified	It is suggested that a clear explanation of the classification of internal/external hazards as DBA or DEC be provided in GD-337.

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			as DBA or DEC. This means that internal and external events can be considered either design basis (if classified AOO or DBA) or complementary design features (if classified as DEC). The criteria for classification of internal/external hazards as DBA or DEC are not addressed in GD-337.	
23	7.3		Since Figure 1 in Section 7.2 of draft RD-337 version 2 shows the plant states, it is more appropriate to include it in Section 7.3 of GD-337.	It is suggested that Figure 1 from Section 7.2 of draft RD-337 be added to Section 7.3. It is further suggested that GD-337 include a version of Figure 1 that also shows the design basis and complementary design features against the operational states and accident conditions. It is also suggested that the following statement be added to describe Figure 1: "The relationship between the plant design envelope and the plant states is shown in Figure 1."
24	7.3	"The design should include the following:... • final safe configurations after AOOs, DBAs, and DECs"	Use of Beyond Design Basis Accident is preferred because it is the commonly used term in the Canadian nuclear industry.	Suggest revising the text as follows: "The design should include the following:... • final safe configurations after AOOs, DBAs, and **BDBAs**"
25	7.3.1	"Operating configurations for normal operation are addressed by the OLCs.....These typically include:... • shutdown in a refuelling	Editorial: The text should be rephrased to achieve greater clarity. Also, it would be useful to explicitly identify guaranteed shutdown state as a normal	Suggest revising the text as follows: "Operating configurations for normal operation are addressed by the OLCs.....These typically include:... • "**refuelling or other maintenance**

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		mode or other maintenance condition that opens the reactor coolant or containment boundary…”	operating mode.	**condition that opens the reactor coolant or containment boundary while in a shutdown mode (i.e., Guaranteed shutdown state)…**”
26	7.3.2	“The plant parameters that are important to the outcome of the safety analysis should be identified. These parameters would typically include:... • core temperature...”	The core temperature is not a directly measured plant parameter. The inlet temperature to the core and the average outlet temperature from the core are directly measured plant parameters.	Suggest revising the text as follows: The plant parameters that are important to the outcome of the safety analysis should be identified. These parameters would typically include:... • “core temperature **(based on the difference between measured core inlet and core outlet temperatures)**”
27	7.3.2	“The plant parameters that are important to the outcome of the safety analysis should be identified. These parameters would typically include:... • temperatures and flows…”	Editorial: The text should be rephrased to achieve greater clarity.	Suggest revising the text as follows: “The plant parameters that are important to the outcome of the safety analysis should be identified. These parameters would typically include:... • “temperatures and flows **for process systems involved in the PIEs**”
28	7.3.4	Discussion of the term “**Design Extension Conditions**” throughout this section.	Use of the term BDBAs is preferred.	Suggest revising the text to discuss BDBAs rather than DECs.
29	7.3.4		Section 7.3.4 of draft RD-337 version 2 states “**The design shall be such that plant states that could lead to significant radioactive releases are practically eliminated; if not, only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public, and sufficient time shall be made available to implement**	It is suggested that further clarification regarding the term “practically eliminated” be provided in Section 7.3.4. It is suggested that further clarification be provided regarding the phrase “only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public”. If applicable, it

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			these measures." GD-337 defines "practically eliminated" in the Glossary, but does not make reference to the term in the body of the document. The use of the term "practically eliminated" requires further clarification. This clarification is not provided in GD-337. The use of the phrase "only protective measures that are of limited scope in terms of area and time shall be necessary for protection of the public" requires further clarification. Is this phrase intended to make reference to the use of sheltering, evacuation and relocation? If so, it is suggested that the text be revised to be consistent with the idea of "implementation of offsite emergency measures".	is suggested that the text be revised to be consistent with the idea of "implementation of offsite emergency measures".
30	7.3.4	"Accidents in this category are, typically, sequences involving more than one failure....The analysis of those accidents may:.... • take credit for realistic system action and performance beyond original intended functions, including systems not important to safety"	Editorial: The text should be rephrased to achieve greater clarity with respect to the definition of "realistic system action and performance beyond original intended functions". It is suggested that the term "physically possible" replace the term "realistic" in order to better communicate the intent. Nevertheless, there is a need for greater clarity on the principles and guidelines to use when analyzing design extension conditions.	Suggest revising the text as follows: "Accidents in this category are, typically, sequences involving more than one failure....The analysis of those accidents may:.... • take credit for **physically possible** system action and performance beyond original intended functions, including systems not important to safety"

#	Section	Excerpt of Section	Industry Issue	Suggested Change
31	7.3.4.1	"Detailed analysis should be performed and documented to identify and characterize accidents that can lead to significant core damage or offsite releases of radioactive material (severe accidents)."	This statement does not consider BDBAs for the spent fuel bays that include postulated significant fuel damage.	Suggest revising the text as follows: "Detailed analysis should be performed and documented to identify and characterize accidents that can lead to significant **core/fuel** damage or offsite releases of radioactive material (severe accidents)."
32	7.6.1		To provide guidance on the requirement in Section 7.6.1 of draft RD-337 version 2, it is suggested that the following text be moved from RD-337 to GD-337: "Failure of a number of devices or components to perform their functions may occur as a result of a single specific event or cause. Common-cause failures may also occur when multiple components of the same type fail at the same time. This may be caused by occurrences such as a change in ambient conditions, saturation of signals, repeated maintenance error or design deficiency."	Suggest adding the following text (originally from Section 7.6.1 of draft RD-337 version 2) to GD-337: "Failure of a number of devices or components to perform their functions may occur as a result of a single specific event or cause. Common-cause failures may also occur when multiple components of the same type fail at the same time. This may be caused by occurrences such as a change in ambient conditions, saturation of signals, repeated maintenance error or design deficiency."
33	7.6.1.2	"The design should implement adequate diversity in safety systems, such as:.. • human diversity"	Editorial: The text should be revised to achieve greater clarity.	Suggest revising the text as follows: "The design should implement adequate diversity in safety systems, such as:.. • human factor engineering diversity"
34	7.6.2		Draft RD-337 version 2 states "2. all identifiable but non-detectable failures, including those in the non-tested components". The inclusion of identifiable, but non-detectable failures, including those in non-tested components appears to exceed the	If it is decided that the requirement regarding "all identifiable but non-detectable failures, including those in non-tested components" is not going to be deleted from RD-337 (as suggested in the comments provided for draft RD-337 version 2), then it is suggested that additional clarification on the expectations

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			definition and intent of "single failure criterion", as described in IAEA Specific Safety Guide SSG-2, Deterministic Safety Analysis for Nuclear Power plants. In the comments provided for draft RD-337 version 2, it was suggested that this requirement be deleted. If it is decided that this requirement will not be deleted, then additional clarification on the expectations for meeting this requirement should be provided in GD-337.	for meeting this requirement be provided in GD-337.
35	7.9.2	"The standards and codes used for computer-based systems or equipment are identified prior to the design."	There are no codes applied for computer-based systems and equipment, only standards. Therefore it is suggested that "codes" be replaced with "practices" in order to be consistent with draft RD-337 version 2.	Suggest revising the text as follows: "The standards **and practices** used for computer-based systems or equipment are identified prior to the design."
36	7.9.2	"The verification and validation activities should be identified and use a top-down approach."	Verification testing is generally performed using a bottom-up approach (e.g., unit test and then subsystem/integration testing). Therefore a bottom- up approach should also be allowed and recognized.	Suggest revising the text as follows: "The verification and validation activities should be identified and use appropriate engineering approaches, e.g., either a top-down or bottom-up approach."
37	7.9.2	"The relationship between design and verification and validation should be indicated and the outcome of verification and validation activities should be documented. The relationship between lifecycle and verification and validation activities should be stated."	Editorial: Improved clarity is needed for "The relationship between lifecycle and verification and validation activities should be stated." Lifecycle consists of design, verification and validation activities.	Suggest revising the text as follows: "The relationship between design and verification and validation should be indicated and the outcome of verification and validation activities should be documented. **The lifecycle should identify when design verification and validation activities are performed in relation to the stages in the design processes**."
38	7.10	"Pre-installed equipment can be credited after 30 minutes where	The basis and justification for changing from an Industry standard of 15 minutes for	Suggest revising the text as follows: "Pre-installed equipment can be credited

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		only control room actions are needed or after 1 hour if field actions are needed.”	operator action in the control room and 30 minutes for operator action outside of the control needs to be provided. This change does not appear to be consistent with IAEA guidance.	after **15 minutes** where only control room actions are needed or after **30 minutes** if field actions are needed.”
39	7.13.1	“Design and beyond design load categories are defined to demonstrate structural performance in operational states and accident conditions.”	Editorial: The text should be revised to achieve greater clarity. In particular, the different types of accident conditions should be addressed.	Suggest revising the text as follows: “Design load categories are defined to demonstrate structural performance in operational states and design basis accident conditions. In addition, beyond design load categories are considered for structural performance in design extension conditions.”
40	7.13.1	“….CSA N289.3-10, *Design procedures for seismic qualification of nuclear power plants*, clause 5.2.2”	Editorial: Clause 5.2.2 should be replaced with clause 5.2.3.	Suggest revising the text as follows: “…..CSA N289.3-10, *Design procedures for seismic qualification of nuclear power plants*, clause **5.2.3**”
41	7.13.1	“Damping ratios for structural systems and sub-systems should be taken into account according to ASCE 43-05.”	The guidance should not be restricting the use of damping ratios to just ASCE 43-05. The damping ratio in CSA N289.3-2010 Table 4 should also be allowed.	Suggest revising the text as follows: Damping ratios for structural systems and sub-systems should be taken into account according to **recognized standards such as ASCE 43-05 and CSA N289.3.**”
42	7.22.3 Table 1	“Ductility ratios”	Editorial: Clarification is needed to explain that the values of ductility ratios in Table 1 are the same for both DBT/DBA and BDBT/BDBA conditions.	Suggest adding the following note to Table 1: “These ductility ratios are equally applicable for DBT/DBA and BDBT/BDBA conditions.”
43	7.22.3 Table 1	“Ductility ratios and supporting rotations”	Editorial: It needs to be clarified whether both the ductility ratios and support rotations shall be met at the same time, as specified in CSA S850-12 (i.e., it fails when either of the ductility ratio or first tier BDBT rotation or second tier BDBT rotation exceeds its corresponding criteria).	Suggest adding the following note to Table 1: “The ductility ratios and support rotations shall be met at the same time, as specified in CSA S850-12 (i.e., it fails when either of the ductility ratio or first tier BDBT rotation or second tier BDBT rotation exceeds its corresponding criteria).”

#	Section	Excerpt of Section	Industry Issue	Suggested Change
44	7.22.3 Table 1	"Support rotations for DBT"	It is unclear how to design SSCs being "essentially elastic." In Note (6), the strain 1% for reinforcement implies the steel bars are much more beyond yield point; and 0.35% concrete compression strain means over concrete peak strength point and is almost crushed. This does not seem to correspond to the elastic response of reinforced/pre-stressed structures/members. Clarification is needed.	Suggest providing clarification for Note (6) or revising Note (6).
45	7.22.3 Table 1	"Failure criteria for DBT"	Since "essentially elastic" response is not a specific rotation, it is hard to directly use it in the design process. Using the support rotation in the DBT column cannot provide insight to engineers in design against DBA/DBT events. It is suggested that the DBT column be removed since it will be automatically governed by the ductility ratio for this condition. The ductility ratios such as those in CSA N287.3 or ACI 349-06 are well developed for application to DBA events. Thus, for DBT conditions, the current ductility criteria should be used.	Suggest deleting the DBT column from Table 1.
46	7.22.3 Table 1	"Support rotations for BDBT"	Clarification is needed for when the UFC 3-340-02 criteria apply to nuclear containment structures with controllable leak tightness. The support rotations are based on the experimental results of the concrete members, which might have significantly different cross sections compared to those in nuclear civil structures.	Suggest adding further clarification to Table 1 regarding the use of the criteria for support rotations for BDBT.

#	Section	Excerpt of Section	Industry Issue	Suggested Change
47	7.22.3 Table 1	“BDBT support rotations for shell-type containment”	Clarification is needed regarding the definition of the term “support rotation” for various types of structures such as dome or cylindrical shells. For various types of containment structures, the criteria for support rotations may be easier to apply to beam/column/wall-panel members, when simplified as SDOF systems as described in CSA S850-12.	Suggest adding text to clarify the CNSC expectations for “support rotation” for various types of structures such as dome or cylindrical shells.
48	7.22.3 Table 1	“BDBT acceptance criteria”	Use of permissible strain limits in the nonlinear 3D finite element analyses, such as in the analysis of Ultimate Pressure Capacity (UPC), provides practical engineering rules. From some test results for nuclear containments, the permissible strain limits specified in US NRC RG 1.216 and/or NUREG/CR-6906 may be applicable to the BDBT events for the corresponding loading conditions.	Suggest adding text to allow for alternative BDBT failure acceptance criteria to facilitate practical analysis and design against blast and impact loading on civil structures in nuclear industry.
49	7.22.3 Table 2	“Failure criteria of steel reinforcement for concrete structures”	Table 2 specifies permissible strains for reinforced steel and post-tensioning steel. Clarification is needed on the use of the criteria for the permissible strains of reinforcing steel in Table 2 with respect to the ductility ratios and support rotations in Table 1.	Suggest adding notes to Table 2 to provide clarification regarding the relationship between the acceptance criteria in Tables 1 and 2.
50	7.22.3 Table 2	“Steel failure criteria”	Due to the nature of impact and impulsive loading, the steel allowable strains based on NEI 07-13 may be applicable, but these values are significantly greater than those from Sandia tests for UPC. The reason for the differences are likely due to the dynamic versus static responses to the impact and impulsive loadings.	The rationale for the suggested values to be applied in design should be included.

#	Section	Excerpt of Section	Industry Issue	Suggested Change
51	8.1.0.3	"The reactor internal components designated as ASME Code, Section III, *Core Support Structures* should be designed, fabricated, and examined in accordance with the provisions of Section III, subsection NG, of the ASME Code."	The terminology used in this statement is not in accordance with the ASME Code. It should be noted that subsection NG of the code does not apply to components (refer to ASME definition of component in NCA-9000); it applies to core support structures and internal structures. The suggested change is in accordance with the ASME terminology.	Suggest revising the text as follows: "The reactor **internals classified as Core Support Structures according to ASME BPVC Section III Division 1 NG-1121,** should be designed, fabricated, and examined in accordance with the provisions of **ASME BPVC Section III Division 1, subsection NG**."
52	8.1.0.3	"Those reactor internals components not designated as ASME Code, Section III, *Core Support Structures* should be designated as internal structures in accordance with ASME Code, Section III, Subsection NG-1122. The design criteria, loading conditions, and analyses that provide the basis for the design of reactor internals (other than the core support structures) should meet the guidelines of ASME Code, Section III, Subsection NG-3000, and constructed so as to not adversely affect the integrity of the core support structures. If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria,	The terminology used in this paragraph is not in accordance with the ASME Code. It should be noted that Subsection NG of the code does not apply to components (refer to ASME definition of component in NCA-9000); it applies to core support structures and internal structures. Please refer to ASME BPVC Section III, NG-1121 and NG-1122 for definitions of core support structures and internal structures, and the applicability of the NG subsection to both of these structures. The suggested change is in accordance with the ASME terminology.	Suggest revising the text as follows: "**For those reactor internals classified as internal structures in accordance with ASME Code, Section III, Division 1, Subsection NG-1122, the design criteria, loading conditions, and analyses that provide the basis for their design requirements should meet the guidelines of ASME Code, Section III, Division 1, Subsection NG-3000, and they should be constructed so as not to adversely affect the integrity of the core support structures**. If other guidelines (e.g., manufacturer standards or empirical methods based on field experience and testing) are the bases for the stress, deformation, and fatigue criteria, those guidelines should be identified and their use justified in the design."

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		those guidelines should be identified and their use justified in the design.”		
53	8.1.0.3	“For non-ASME code structures and components, design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. Any decreases in design margins should be justified.”	This sentence should be applicable to reactor internals other than those which the ASME code covers (i.e. anything other than pressure retaining components or supports, core support structures and internal structures). Supports have not been addressed in this sentence.	Suggest revising the text as follows: “For non-ASME code **structures, components and supports**, design margins presented for allowable stress, deformation, and fatigue should be equal to or greater than margins for other plants of similar design with successful operating experience. Any decreases in design margins should be justified.”
54	8.1.0.3	“Specific reactor internals components designated as Class 1, Class 2, and Class 3 should be designed, fabricated, and examined in accordance with the applicable codes and standards, such as ASME Section III for light water reactors (LWR), and CSA N285.0, *General requirements for pressure-retaining systems and components in CANDU nuclear power plants* for CANDU.”	This paragraph should be revised in accordance with ASME terminology. It should be noted that Subsection NG of the code does not apply to components (refer to ASME definition of component in NCA-9000); it applies to core support structures and internal structures. It is further suggested that this paragraph be moved to the beginning of the subsection.	Suggest moving this paragraph to the beginning of the subsection and revising the text as follows: “Specific reactor **internal or core support structures classified as Class 1, Class 2, and Class 3 in accordance with ASME BPVC Section III Division 1, Subsection NCA-2000**, should be designed, fabricated, and examined in accordance with the applicable codes and standards, such as ASME **BPVC** Section III for light water reactors (LWR), and CSA N285.0, *General requirements for pressure-retaining systems and components in CANDU nuclear power plants* for CANDU.”
55	8.2	“For designs that include a pressurizer, the design authority should demonstrate the adequacy of the following:… • control of pressure via	Pressure can also be controlled by steam bleeding.	Suggest revising the text as follows: “For designs that include a pressurizer, the design authority should demonstrate the adequacy of the following:… • control of pressure via heaters, sprays,

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		heaters, sprays or coolers”		coolers or **steam bleeding**”
56	8.4		For LWRs, a control rod ejection is a possible postulated initiating event. The text should include guidance on the means of shutdown to account for this type of event.	It is suggested that this section be revised to provide guidance on the means of shutdown to account for possible control rod ejection.
57	8.6.12	Discussion of the term “**Design Extension Conditions**” throughout this section.	Use of the term BDBAs is preferred.	Suggest revising the text to discuss BDBAs rather than DECs.
58	8.6.12	“Containment leakage rate in DECs does not exceed the design leakage rate for sufficient period to allow for the implementation of offsite emergency measures.”	It should be clarified that this requirement only applies to DECs with core damage.	Suggest revising the text as follows: “Containment leakage rate in DECs **with core damage** does not exceed the design leakage rate for sufficient period to allow for the implementation of offsite emergency measures.”
59	8.9.1	**Station blackout** “A complete loss of alternating current (AC) power from offsite and onsite main generator, standby and emergency power sources. Note that it does not include failure of uninterruptible AC power supplies (UPS) and DC power supplies. It also does not include failure of alternate AC power.”	It is suggested that some additional clarification is needed to accompany the definition of station blackout. To achieve greater clarity, the complete loss of AC power from offsite and onsite main generator, standby and emergency power sources needs to be defined as: - the loss of supply of AC power to essential and non-essential switchgear buses in a nuclear power plant, - the unavailability of standby and emergency power sources that automatically start up and connect in response to the loss of offsite power and a turbine trip, - excluding a concurrent single failure, and - excluding a concurrent design basis	Suggest revising the text to provide additional clarification. ”

#	Section	Excerpt of Section	Industry Issue	Suggested Change
			accident. Furthermore, it is suggested that the definition of station blackout should exclude assumptions of failure to standby AC power sources that are dedicated to powering SSCs that are complementary design features, provided the applicable requirements are met.	
60	10.1	"The design should incorporate the "best available technology and techniques economically achievable" (BATEA) principle for aspects of the design related to environmental protection."	The introduction of the term "best available technology and techniques economically achievable" goes beyond the current Canadian environmental protection regulations. This is introducing new requirements that may not be consistent with the current Canadian Environmental Protection Act.	Suggest deleting this statement.
61	10.2	"The design authority should demonstrate adherence to the principles of optimization and pollution prevention, through the demonstration of the application of ALARA and BATEA principles."	The introduction of the term "best available technology and techniques economically achievable" goes beyond the current Canadian environmental protection regulations. This is introducing new requirements that may not be consistent with the current Canadian Environmental Protection Act.	Suggest revising as follows: "The design authority should demonstrate adherence to the principles of optimization and pollution prevention, through the demonstration of the application of ALARA principles."
62	Glossary		Add definition of "proven design" from draft RD-337 version 2.	Suggest adding the following term to the glossary: "**proven design** A design of a component(s) can be proven either by showing compliance with accepted engineering standards, or by a history of experience, or by test, or some combination

#	Section	Excerpt of Section	Industry Issue	Suggested Change
				of these. New component(s) are "proven" by performing a number of acceptance and demonstration tests that show the component(s) meets pre-defined criteria."
63	Glossary	"**anticipated operational occurrence** An operational process deviating from normal operation, which is expected to occur at least once during the operating lifetime of a facility, but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions."	The definition of anticipated operational occurrences is not identical to that provided in the glossary in RD-310. Consistency is required.	Suggest revising the definition in this document to be consistent with that provided in RD-310: "**anticipated operational occurrence** An operational process deviating from normal operation that is expected to occur once or several times during the operating lifetime of the NPP but which, in view of the appropriate design provisions, does not cause any significant damage to items important to safety nor lead to accident conditions."
64	Glossary	"**cliff-edge effect** A large increase in the severity of consequences caused by a small change of conditions. Note: cliff-edges can be caused by changes in the characteristics of the environment, the event or changes in the plant response."	The impact of this proposed wording requires further evaluation, particularly in light of the work and projects in progress to meet RD-310 requirements. Therefore the term "cliff edge effects" should not be used.	It is suggested that this term be deleted from GD-337 pending further evaluation.
65	Glossary	"**complementary design feature** A design feature added to the design as a stand-alone structure, system or component (SSC) or added capability to an existing SSC to cope with design extension conditions."	See comment #21	Suggest providing clarification on positioning portable equipment under systems important to safety in complementary design features for new nuclear power plants.
66	Glossary	"**mission time**	Editorial: For clarity, suggest adding "safety"	Suggest revising the text as follows:

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		The duration of time within which a system or component is required to operate or be available to operate and fulfill its function following an event.”	before “function” and allowing for multiple safety functions.	“**mission time** The duration of time within which a system or component is required to operate or be available to operate and fulfill its **safety** function**(s)** following an event.”
67	Glossary	“**probabilistic safety assessment** A comprehensive and integrated assessment of the safety of the nuclear power plant. 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 nuclear power plant, as follows: 1. 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 2. a Level 2 PSA starts from the Level 1 results and analyses the containment behaviour, evaluates the radionuclides released from the failed fuel and quantifies the releases to	The definition of **probabilistic safety assessment** is not identical to that provided in the glossary in S-294. Consistency is required.	Suggest revising the definition in this document to be consistent with that provided in S-294: “**probabilistic safety assessment** For a NPP or a fission nuclear reactor, a comprehensive and integrated assessment of the safety of the plant or reactor. 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 plant or reactor, as follows: 1. 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 2. a Level 2 PSA starts from the Level 1 results and analyses the containment behaviour, evaluates the radionuclides released from the failed fuel and quantifies the releases to the environment 3. a Level 3 PSA starts from the Level 2 results and analyses the distribution of radionuclides in the environment and evaluates the resulting effect on public health.

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		the environment 3. a Level 3 PSA starts from the Level 2 results and analyses the distribution of radionuclides in the environment and evaluates the resulting effect on public health.”		A PSA may also be referred to as a Probabilistic Risk Assessment (PRA).”
68	Glossary	“**severe accident** Accident conditions more severe than a design basis accident and involving significant core degradation.”	As written, the definition of severe accident does not encompass beyond design basis accidents involving the spent fuel bay where significant fuel degradation would be a postulated scenario. Suggest replacing “significant core degradation” with “significant fuel degradation” to encompass BDBAs for the spent fuel bay. This change would not have an impact on the intent of the definition of severe accident when applied to the reactor core. A change to the definition is also needed to make it consistent with Section 7.3.4.1, “Severe accidents represent accident conditions that involve significant fuel degradation, either in-core or in-fuel storage.”	Suggest revising the text as follows: “**severe accident** Accident conditions more severe than a design basis accident and involving significant **fuel** degradation.”
69	Glossary	“**shutdown state** A state characterized by subcriticality of the reactor. At shutdown, automatic actuation of safety systems could be blocked and support systems may remain in abnormal	Replace “actuation of safety systems could be blocked” to “actuation of safety systems **may** be blocked”. This suggestion is to make the definition consistent with the use of “may” and “can” from the preface.	Suggest revising the text as follows: “**shutdown state** 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

#	Section	Excerpt of Section	Industry Issue	Suggested Change
		configurations.”	Any blocking of safety system actuation is only permissible within the limits of the regulatory requirements.	configurations.”
70	Glossary	“**station blackout** A complete loss of alternating current (AC) power from offsite and onsite main generator, standby and emergency power sources. Note that it does not include failure of uninterruptible AC power supplies (UPS) and DC power supplies. It also does not include failure of alternate AC power.”	Suggest identifying this is also “extended loss of AC power event” – consistent with use of term in industry.	Suggest revising the text as follows: “**station blackout** (also known as extended loss of AC power event) A complete loss of alternating current (AC) power from offsite and onsite main generator, standby and emergency power sources. Note that it does not include failure of uninterruptible AC power supplies (UPS) and DC power supplies. It also does not include failure of alternate AC power.”
71	Glossary	“**ultimate heat sink** A medium to which the residual heat can always be transferred and is normally an inexhaustible natural body of water or the atmosphere.”	Suggest using the IAEA definition, rather than paraphrasing the IAEA definition.	Suggest revising the text as follows: “**ultimate heat sink** A medium into which the transferred residual heat can always be accepted, even if all other means of removing the heat have been lost or are insufficient. This medium is normally a body of water or the atmosphere.”