Bruce Power Comments on GD-337, Guidance on the Design of New Nuclear Power Plants

From: BOYADJIAN Joe(J) - BRUCE POWER [mailto:joe.boyadjian@brucepower.com]
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Subject: Bruce Power Comments on GD-337, Guidance for the Design of New Nuclear Power Plants
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In response to Reference 1, attached find please Bruce Power's comments on GD-337, Guidance for the Design of New Nuclear Power Plants.

If you have any questions or require further information on this submission, please contact Mr. Maury Burton, Department Manager, Regulatory Affairs at 519-361-2673 extension 15291 or at maury.burton@brucepower.com.

Sincerely;

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Reference:

1. Information Bulletin 12-42**,** Subject: Invitation to Comment on Draft GD-337, Guidance for the Design of New Nuclear Plants

	Document Section/Excerpt	Industry Issue	Suggested Change
1.	**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.	Update GD-337 after RD-337 has been finalized and approved, and then issue it again for public consultation.
2.	**General**	The CNSC should take into consideration comments submitted on RD-337 for revisions to GD-337.	Use comments provided during the public consultation phase of RD-337 to update GD-337.
3.	**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 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. Also, the CNSC should use consistent terminology for DEC in RD-337; consistency with Section 7.3, 4.2.3 and definitions provided in glossary are needed.	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 extension conditions". The CNSC should also provide guidance on the principles and guidelines for performing deterministic safety analyses for "design extension conditions".
4.	**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 this practice be carried forward for other GDs & RD/GDs
5.	**General**	Many standards with the edition dates are referenced throughout the document. This is not a good practice, because newer editions of the standards will be issued between revisions to GD-337.	It is suggested that the edition dates not be included or to included a statement regarding the use of more recent editions of the standards.
6.	***Preface and Section 2*** "SSR 2/1, *Safety of Nuclear Power*	Editorial: The correct title of SSR-2/1 is "Specific Safety Requirements: Safety	Suggest title of the document be corrected to:

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	Plants: Design"	of Nuclear Power Plants: Design"	"… SSR-2/1, **Specific** Safety Requirements: Safety of Nuclear Power Plants: Design"
7.	**Section 3 Bullet 5**	The list of clauses 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 clauses 5(k), 6(j) and 6(k), however these clauses are not listed.	Suggest that final version of GD-337 be reviewed against the Class I Nuclear Facilities Regulations for completeness.
8.	**Section 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 safety systems", 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.	
9.	**Section 5.3** • 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 of design documentation and records	The bullets do not follow a "chronological" order. The design control measures listed here should follow in order how the design activities progress from initiation to being ready for implementation, as described in CSA N286-05. Also note that CSA N286 June 2012 has been issued and may supersede CSA N286-05. Some bullets are partially included in other bullets. As example, planning of design activities is mentioned in both 1st and 4th bullets. The bullet "management of the design and control of design changes" is included in the bullet "configuration management". The bullet "conducting conceptual analysis" should be more specific about	Suggest changing the text to: "• design initiation, including identification of scope • work control and planning of design activities • selection 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

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	• 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"	the type of analysis (safety, stress??). CSA N286 clearly indicates a conceptual safety analysis to assess the preferred design concept. The bullet "selection of suitably qualified and experienced staff" may suggest that only experienced staff can perform design activities, while CSA N286-05 requirement is for personnel competent to do the design work assigned to them (competence includes education, training, skills, experience and ability). It is suggested that all bullets in GD section 5.3 follow CSA N286-05.	production of design documentation and records • conducting detailed safety analysis to prove adequacy of detailed design • defining any limiting conditions for safe operation • carrying out design verification and validation • configuration management • identification and control of design interfaces"
10.	**Section 5.3**	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. 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".	Suggest adding the following text: "As stated in RD-337, "The computer software used for design and analysis calculations shall be qualified in accordance with applicable standards. This is achieved by following industry standards for software, such as CSA N286.7, where qualified software: (a) is shown to be capable of addressing 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

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			and change control."
11.	**Section 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 more properly belongs at the end of Section 6.1, rather than at the end of Section 6.1.1. Section 6.1.1 is about the physical barriers, whereas this paragraph is applicable to the design features for all levels of defence-in-depth.	Suggest moving this paragraph to the end of Section 6.1.
12.	**Section 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. It is not clear that a greater emergency response capability is necessary for a larger exclusion zone.	Suggest changing the text to: "Generally, a larger exclusion zone would allow for more emergency response time."
13.	**Section 6.5** Evacuation Needs	Environmental factors also affect evacuation times (precipitation = slower evacuation). This is not specifically mentioned here, although consideration of this usually appears in the nuclear emergency response plans.	Suggest adding the following text: "Environmental factors which can affect the response times should be taken into consideration."
14.	**Section 6.6.1** "As stated in 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.	Suggest changing all use of :multi-unit site" to "multiple units at a site".
15.	**Section 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 changing the text to: "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."
16.	**Section 7.1** "The SSC classification process should include the following activities: • identification of engineering design	The SSC classification process should not include the identification of engineering design rules for classified SSCs. Once a safety class has been	Suggest changing the text by replacing the bullet "identification of engineering design rules for classified SSCs" with the following paragraph:

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	rules for classified SSCs"	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.	"Once the safet class of SSCs isestablished, 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 its 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."
17.	**Section 7.1** "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 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 changing the text to" "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"
18.	**Section 7.1** "Although the probability of SSCs being called upon during DECs is very low,	The phrase "these safety functions should be considered a high safety category" needs clarification. The term	Suggest changing the text to: "Although the probability of SSCs being

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	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."	"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 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.	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 significance."
19.	**Section 7.1** "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.
20.	**Section 7.1**	RD-337 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. This additional clarification should be included in GD-337.	

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21.	**Section 7.2**	The criteria for classification of internal/external hazards as DBA or DEC are not clearly explained in GD-337.	
22.	**Section 7.3**	Since Figure 1 of RD-337 version 2 shows the plant states, it is more appropriate to include it in Section 7.3 of GD-337. It is also suggested that GD-337 could include a version of Figure 1 that also shows the design basis and complementary design features against the operational states and accident conditions.	Suggest adding the following text to Section 7.3 GD-337 along with Figure 1 from RD-337 version 2: "The relationship between the plant design envelope and the plant states is shown in Figure 1."
23.	**Section 7.3.1** "shutdown in a refuelling mode or other maintenance condition that opens the reactor coolant or containment boundary"	Editorial: The text needs rephrasing to achieve greater clarity. Also, it would be useful to explicitly identify guaranteed shutdown state as a normal operating mode.	Suggest changing the text to: • "refuelling or other maintenance condition that opens the reactor coolant or containment boundary while in a shutdown mode • Guaranteed shutdown state"
24.	**Section 7.3.2** "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 changing the text to: "core temperature (based on the difference between measured core inlet and core outlet temperatures)"'
25.	**Section 7.3.2** "temperatures and flows"	Editorial: The text needs rephrasing to achieve greater clarity.	Suggest changing the text to: "temperatures and flows for process systems involved in the PIEs"
26.	**Section 7.3.4**	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	

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		made available to implement 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 text should be revised to put it into context with respect to meeting the safety goals. 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 changed to be consistent with the idea of “implementation of offsite emergency measures”.	
27.	**Section 7.3.4** “take credit for realistic system action and performance beyond original intended functions, including systems not important to safety”	Editorial: The text needs rephrasing to achieve greater clarity with respect to the definition of “realistic system action and performance beyond original intended functions”. Perhaps using “physically possible” rather than “realistic” can communicate the intent better, Nevertheless, there is a need for greater clarity on the principles and guidelines to use when analyzing design extension conditions.	Suggest changing the text to: “take credit for physically possible system action and performance beyond original intended functions, including systems not important to safety”
28.	**Section 7.6.1**	To provide guidance on the requirement in Section 7.6.1 of RD-337	Suggest adding the following text:

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		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."	"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."
29.	**Section 7.6.1.2** "human diversity"	Editorial: The text needs rephrasing to achieve greater clarity.	Suggest changing the text to: "human factor engineering diversity"
30.	**Section 7.6.2**	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 definition and intent of "single failure criterion", as described in IAEA Specific Safety Guide SSG-2, Deterministic Safety Analysis for Nuclear Power plants. If this requirement is not removed from RD-337, then additional clarification on the expectations for meeting this requirement is needed in GD-337.	
31.	**Section 7.9.2** "The standards and codes used for computer-based systems or equipment are identified prior to the design."	Replace codes with practices as per RD-337 version 2, because there are no codes applied for computer-based systems and equipment, only standards.	Suggest changing the text to: "The standards and practices used for computer-based systems or equipment are identified prior to the design."

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32.	**Section 7.9.2** “The verification and validation activities should be identified and use a top-down approach.”	A bottom up approach should also be allowed and recognized. Verification testing is generally perform using a bottom-up approach (e.g., unit test and then subsystem/integration testing).	Suggest changing the text to: “The verification and validation activities should be identified and use appropriate engineering approaches, e.g., either a top-down or bottom-up approach.”
33.	**Section 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 changing the text to: “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.”
34.	**Section 7.10** “Pre-installed equipment can be credited after 30 minutes where only control room actions are needed or after 1 hour if field actions are needed.”	The basis and justification for changing from an Industry standard of 15 minutes for 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.	Suggest changing the text to: “Pre-installed equipment can be credited after 15 minutes where only control room actions are needed or after 30 minutes if field actions are needed.”
35.	**Section 7.13.1** “Design and beyond design load categories are defined to demonstrate structural performance in operational states and accident conditions.”	Editorial: The text needs rephrasing to achieve greater clarity.	Suggest changing the text to: “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.”
36.	**Section 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 clause 5.2.3.	Suggest changing the text to: “CSA N289.3-10, *Design procedures for seismic qualification of nuclear power plants,* clause 5.2.3”
37.	**Section 7.13.1**	The guidance should not be restricting	Suggest changing the text to:

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	“Damping ratios for structural systems and sub-systems should be taken into account according to ASCE 43-05.”	the use of damping ratios to just ASCE 43-05. The damping ratio in CSA N289.3-2010 Table 4 should also be allowed.	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.”
38.	**Section 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 a note to Table 1: “These ductility ratios are equally applicable for DBT/DBA and BDBT/BDBA conditions.”
39.	**Section 7.22.3, Table 1** “Ductility ratios and support rotations”	Editorial: Clarification is needed that 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 a 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.”
40.	**Section 7.22.3, Table 1** “Support rotations for DBT”	DBT support rotations: 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 seems not to correspond to the elastic response of reinforced/prestressed structures/members. Please clarify this.	Suggest providing clarification for Note (6) or revising Note (6).
41.	**Section 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 this DBT in the column cannot provide insight to engineers in design against DBA/DBT events. It is suggested to remove this column (DBT) since it will be automatically	Suggest deleting the DBT column from Table 1.

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		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.	
42.	**Section 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.
43.	**Section 7.22.3, Table 1** "BDBT support rotations for shell-type containment"	Clarification is needed on 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.
44.	**Section 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.
45.	**Section 7.22.3, Table 2** "Failure criteria of steel reinforcement for concrete structures"	Table 2 specifies permissible strains for reinforce steel and post-tensioning steel. Clarification is needed on the use of the criteria for the permissible	Add clarification as notes to Table 2 for the relationship between the acceptance criteria in Tables 1 and 2.

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		strains of reinforcing steel in Table 2 with respect to the ductility ratios and support rotations in Table 1.	
46.	**Section 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 significant 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.
47.	**Section 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 is not according to ASME Code. Note that Subsection NG of the code does not apply to components (see ASME definition of component in NCA-9000), applies to core support structures and internal structures. The suggested change is in accordance with the ASME terminology.	Suggest changing the text to: “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.”
48.	**Section 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, those	The terminology is not according to ASME Code. Note that Subsection NG of the code does not apply to components (see ASME definition of component in NCA-9000), applies to core support structures and internal structures. See ASME BPVC Section III, NG-1121 and NG-1122 for definitions of core support structures and internal structures, and applicability of NG subsection to both of them. The suggested change is in accordance with the ASME terminology.	Suggest changing the text to: “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 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.”

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	guidelines should be identified and their use justified in the design.”		
49.	**Section 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 anything else except for what the ASME code covers, which means anything else than pressure retaining components or supports, core support structures and internal structures. Supports were not included in the sentence.	Suggest changing the text to: “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.”
50.	**Section 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.”	Rephrase according to ASME terminology. I suggest to move this paragraph for Class 1/2/3 pressure retaining components and supports at the beginning of the subsection "Reactor internals".	Suggest changing the text to: “Specific reactor internals components or supports 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.”
51.	**Section 8.2** “control of pressure via heaters, sprays or coolers”	Pressure control can also be done by steam bleeding	Suggest changing the text to: “control of pressure via heaters, sprays, coolers or steam bleeding”
52.	**Section 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.	
53.	**Section 8.9.1**	It is suggested that some additional clarification is needed for the definition of station blackout. To achieve greater	

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		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 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.	
54.	**Glossary**	Add definition of "proven design from RD-337 version 2.	Suggest changing the text to: "**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 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."
55.	**anticipated operational occurrence**	The definition of anticipated operational	Suggest revising the definition in this

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	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.	occurrences is not identical to the definition provided in the glossary in RD-310. The definition should be consistent in both documents.	document to be consistent with that provided in RD-310: "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."
56.	**"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 term "cliff edge effects" should not be used. The impact of this proposed wording requires further evaluation, particularly in light of the work and projects in progress to meet RD-310 requirements.	Suggest that this term be deleted from GD-337 pending further evaluation.
57.	**"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."**	For new nuclear power plants, more clarification is required with respect to whether portable equipment should be listed under systems important to safety as complementary design features for new nuclear power plants. For existing nuclear power plants it is noted that portable equipment is not considered to be systems important to safety. This additional clarification should be included in GD-337.	No change to text.
58.	**mission time** 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.	Editorial: For clarity, suggest adding "safety" before "function" and allowing for multiple safety functions.	Suggest changing the text to: "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."
59.	**"probabilistic safety assessment** A comprehensive and integrated	The definition of **probabilistic safety assessment** is not identical to that	Suggest replacing the definition in RD-337 version 2 with the definition

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	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 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.“**	provided in the glossary in S-294. Consistency is required.	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. A PSA may also be referred to as a Probabilistic Risk Assessment (PRA).”
60.	**“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	Suggest changing the text to: “Accident conditions more severe than a design basis accident and involving significant **fuel** degradation.”

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		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."	
61.	"**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 configurations."	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. Any blocking of safety system actuation is only permissible within the limits of the regulatory requirements.	Suggest changing the text to: "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 configurations."
62.	**"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 changing the text to: "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."
63.	**"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 changing the text to: "ultimate heat sink A medium into which the transferred *residual heat* can always be accepted, even if all other means of removing the

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			heat have been lost or are insufficient. This medium is normally a body of water or the atmosphere.”
64.	**All of GD-337**	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 in public review), the combined RD/GD-337 must be clearly structured to differentiate between the requirements that may be used as part of the licensing basis for a regulated facility or activity by reference in a licence and 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 the requirements be identified as “normative” to define the statements as mandatory and the “expectations and guidance” be identified as “informative” to define the statements as a means to meet the requirements.