TITLE: DS508 - Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
1.	France	1	General comment		1. The draft guide is probably not very useful as a practical guide since, as is acknowledged, most aspects need to be treated on a case by case basis and there are a number of different approaches. Understanding these is useful but they do not represent a consensus. The DiD sections add little to the discussion in the Requirements Documents GSR Part 4 and SSR 2/1, which provide adequate high level guidance. 2. There is little guidance given on what constitutes an unacceptable release. The answer seems to be a release which overwhelms the emergency plan. This is discussed in 4.6 for early releases (should be site specific) but not for large releases which are not limited in time or extent. In practice it is the regulator who must decide which is reasonable.			This is neither the intention nor the objective of this safety guide to define what is an unacceptable release since it is the competence of national regulatory authorities.	X	Even though many aspects need to be treated on a case by case basis, general recommendations are still applicable and consensus on them could be reached. Comments have been resolved with this objective. DiD sections have been trimmed following the agreement of the NUSSC WG. Nevertheless, it stills provides guidance on some aspects of DiD implementation that is not provided in the requirements.
2.	ENISS	1	General comment 1		1. The draft guide does not sufficiently support practical use since, as is acknowledged, most aspects need to be treated on a case by case basis and there are a number of different approaches. Understanding these is useful but they do not represent a consensus. The DiD sections add little to the discussion already incorporated in the Requirements Documents GSR Part 4 and SSR 2/1, which provide adequate high-level guidance. 2. There is little guidance given on what constitutes an unacceptable release. The answer seems to be a release which overwhelms the emergency plan. This is discussed in 4.6 for early releases (should be site specific) but not for large releases which are not limited in time or extent. In practice it is the regulator who must decide which is reasonable. 3. While practical elimination by demonstration of extreme unlikelihood is acknowledged to be a probabilistic concept there is no guidance as to what an acceptably low probability might be.					Even though many aspects need to be treated on a case by case basis, general recommendations are still applicable and consensus on them could be reached. Comments have been resolved with this objective. DiD sections have been trimmed following the agreement of the NUSSC WG. Nevertheless, it stills provides guidance on some aspects of DiD implementation that is not provided in the requirements.
3.	UK	1	General		List of abbreviations to be added – terms to be defined on first use (e.g. AOO, DBA in para 1.7)	For clarity/completeness	X			Considered in technical edition

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4.	Canada	1	General **Major comment**		Please re-read the wording of SSR-2/1 paras 5.31 and 5.31A, (together with Footnote 3). As highlighted in the “Reason” column, SSR-2/1 describes two possible outcomes. Protective measures can either be effective (and allowable in DEC), or ineffective (and requiring PE). DS508 paragraphs 2.8 and 4.7 do not require PE until consequences are significantly greater than those permitted in DEC. This creates a new class of accidents with consequences greater than allowed in DEC and less than requiring PE. There is no text in SSR-2/1 that supports this new class of accident. **We believe that accepting the DS508 interpretation will require a change to the wording of SSR-2/1.**	As Canada has commented previously, DS508 effectively introduces a new class of accident more severe than DEC but not so severe that it must be PE. We believe that this violates the requirements of SSR-2/1 Rev. 1. Our comment was rejected for reasons that Canada considers invalid. The issue is argued more fully in the specific comments below. SSR-2/1 requires demonstration that, for DEC, necessary protective actions will be limited in time and area and there will be sufficient time to implement them. It says that large releases must be practically eliminated. A large release is one where protective actions are not limited in time and area. It also says that early releases must be practically eliminated. An early release is one where there is insufficient time to implement protective actions.	Accepted			Text considered in the comment originally in paragraph 2.8, now paragraph 2.6, is deleted. Original paragraph 4.7 is deleted.
5.	Canada	2	General		Review those comments that were rejected in the previous phase.	Canada supports many of the extensive comments made by France on previous draft of DS508 concerning rephrasing the requirements Safety Standards, effectively changing requirements, or even creating new requirements. Also, France made many comments pointing out repetition or material that was out of scope.	Accepted			
6.	Canada	3	General **Major comment**		Restructure the document such that the emphasis on defence in depth is removed. Canada recommends deletion of paragraphs: • 3.31 to 3.51 concerning implementation of DiD. • 3.52 to 3.68 concerning independence between levels of DiD. Some material relevant to DEC and PE could be moved to other parts of the document if it is not there already.	NUSSC approved DPP508 with an objective to provide recommendations on assessment and implementation of new and revised clauses of SSR-2/1 Rev. 1 and GSR Part 4. **The focus requested by NUSSC was the changes relating to DEC and Practical Elimination.** It is misleading to insert partial guidance on defence in depth (DiD) into a document with a different purpose. The contents of DS508 are only relevant to water-cooled NPPs, only address design aspects of DiD and are largely limited to levels 4 and 5 DiD. Therefore, the guidance on DiD is incomplete. A topic as important as DiD that spans all facilities and activities should not be treated in this way.				The consideration of DEC and PE in the frame of DiD was agreed and presented in the DPP. The WG of NUSSC has delimited the areas to be addresses in this safety guide. This comment contradicts these agreements

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7.	Canada	65	General **Major comment**		DS508-step 9 recommends different limits for design extension conditions without significant fuel degradation (DEC-A) and design extension conditions with core melting (DEC-B). SSR-2/1 does not support this. The problem occurs several times in DS508. The phrase "*without significant fuel degradation*" does not occur anywhere in the body of SSR-2/1 (it is in the Glossary section). SSR-2/1 sets requirements for DEC but does not set different requirements for DEC-A and DEC-B. Member States may <u>choose</u> to set more restrictive requirements for DEC-A. SSG-2 para 7.46 makes this clear. An optional position must not be used in an IAEA Safety Guide, though it could be mentioned in a footnote or Annex which are not integral parts of the main text. **It is not acceptable for a Safety Guide to add or recommend limits that exceed the requirements of the Safety Standard.** Dividing DEC into DEC-A and DEC-B is useful when talking about details of analysis such as protection of barriers and key phenomena. However, it leads to circular reasoning as used in DS508. We strongly advise that any optional approaches using differing requirements for DEC-A and DEC-B are carefully worded to avoid circular reasoning. See, for example, the first sentence of 3.48 begins: "*3.48 Safety features for DEC without significant fuel degradation should be demonstrated to be efficient enough according to the applicable analysis rules to prevent core damage for the accident sequences for which they are intended …*" This text effectively says, "*DEC without significant fuel degradation should not lead to significant fuel degradation.*" As a requirement the text is meaningless. Comments are provided for specific paragraphs below.			3.52 The reliability of safety features for design extension conditions without significant fuel degradation should be such that it can be demonstrated, with a sufficient level of confidence and considering applicable analysis rules (see paras 7.45-7.55 of SSG-2 (Rev. 1) [9]), that they are capable of preventing core damage with a frequency higher than the established probabilistic targets.		
8.	Canada	4	1		Consider adding a paragraph (or at least a footnote) to the Background (or section 4) to discuss the meaning of 'practical elimination'. It is problematic that the term has no formal definition beyond a footnote in SSR-2/1. It is also problematic that the footnote is accepted by the IAEA glossary as capable of misinterpretation.	The IAEA Safety Glossary (2018) contains a note concerning common misinterpretations of PE: *"The phrase 'practically eliminated' is misleading as it actually concerns the possible exclusion of event sequences from hypothetical scenarios rather than practicalities of safety. The phrase can also all too readily be misinterpreted, misrepresented or mistranslated as referring to the 'elimination' of 'accidents' by practical measures (or else 'practically' in the sense of 'almost'). Clear drafting in natural language would be preferable.*" DS508 could provide clarification if a common understanding could be agreed.		The leading document for such a topic cannot be the Safety Glossary A proposal of definition is presented in the Definition section, based on the discussions during September meeting.		
9.	Italy	2	1.1	6	[…] operational experience and insights from safety research.	Typo (two "." at the end of the paragraph)	X			Paragraph modified after technical edition
10.	France ENISS	2 2	1.1		Over the latest decades, IAEA safety standards for nuclear power plant design have been enhanced several times with the aim of providing confidence that the successive generations of nuclear power plants are	"nuclear events" is an undefined ambiguous term and the events of interest, including TMI, Chernobyl and Fukushima, are covered by "operational experience" so it can be deleted. Delete repeated full stop at end.	X	Paragraph deleted.		

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					designed so as to operate efficiently at the highest levels of safety that can be reasonably achieved considering the state of the art practices and techniques in science and technology and taking into account the feedback gained from the ~~nuclear events,~~ operational experience and insights from safety research.~~.~~					
11.	Germany	1	1.2	Line 10	… The incorporation of these aspects in the new NPP designs requires specific guidance for the design and the necessary safety assessment. Although specific guidance is provided in safety guides for the design of safety features related to these aspects, overarching guidance on their application to the plant design and on their safety assessment is necessary in a single safety guide.	Clarification; not only application but safety assessment is an important issue as well			X	Original text deleted
12.	Italy	1	1.2	5	[…] aspects in the new nuclear power plant (NPP) designs […]	The acronym NPP has not been introduced before	X			Paragraph modified after technical edition
13.	UK	2	1.2, 2.6		Include footnotes 3&4 of SSR 2/1 (definition of early/large release and of PE), either as footnotes to 2.6 3rd bullet, or as a new paragraph after 2.6.	For clarity Early in the document there needs to be a definition of these key terms. These are referred to throughout Sections 1-3, but are only defined later in paragraph 4.1	X			
14.	UK	3	1.2		Change to "….overarching guidance on the application of a) to c) as part of the plant design is necessary in a single safety guide."	This guide is not about the design of safety measures, but about how DiD, PE etc should be considered for the design – note that para 1.10 correctly states that "The guide is not intended to provide specific recommendations for the design of safety features for DEC……"			X	Original text deleted since it is not appropriate for the justification of the SG.
15.	Germany	2	1.3		IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment of Facilities and Activities [2], also revised after the Fukushima Daiichi accident ~~[2]~~, establishes requirements for safety assessment covering the whole lifetime of all types of facilities and activit~~y~~ies. However, r~~R~~equirements for safety assessment of the design in this publication are not sufficiently detailed for nuclear power plants. ~~However~~, S~~s~~pecific requirements for safety assessment and safety analysis of the design of nuclear power plants are established in SSR-2/1 (Rev. 1) [1], and these need to be considered to address specific requirements in SSR-2/1 (Rev. 1) [1] that are related to measures for strengthening the implementation of the concept of defence in depth focussing on design	Clarification		1.2 The incorporation of these aspects into designs of new nuclear power plants will affect the necessary safety assessment. IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment of Facilities and Activities [3] establishes requirements for performing the safety assessment for all types of facility and activity, including assessment of defence in depth. Specific requirements for safety assessment and safety analysis of nuclear power plants are established in SSR 2/1 (Rev. 1) [1].		

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					extension conditions, mentioned above (Para. 1.2), namely those related to defence in depth and practical elimination of event sequences leading to early radioactive releases or large radioactive releases ~~aspects of relevance for nuclear power plant design~~.					
16.	UK	4	1.3		Change to: IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment of Facilities and Activities, also revised after the Fukushima Daiichi accident [2], establishes requirements for safety assessment covering the whole lifetime of all types of facilities and activity. However, it does not provide detailed requirements specifically for the safety assessment of NPP designs nuclear power plants. SSR-2/1 (Rev. 1) [1] does establish requirements for the safety assessment and safety analysis of NPPs, and these need to be considered to address specific aspects of relevance for nuclear power plant design	The meaning of the text "Requirements for safety assessment of the design in this publication are not sufficiently detailed for nuclear power plants" is not clear. It reads like a self-criticism of DS508 in terms of what it fails to do. It is presumed it is trying to say it is beyond the scope of GSR Part 4 to provide guidance on all aspects of NPP safety assessment.		1.2 …IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment of Facilities and Activities [3] establishes requirements for performing the safety assessment for all types of facility and activity, including assessment of defence in depth. Specific requirements for safety assessment and safety analysis of nuclear power plants are established in SSR 2/1 (Rev. 1) [1]. See also para.1.3 1.3 The objective of this Safety Guide is to provide recommendations for the design of new nuclear power plants on the application of selected requirements in SSR-2/1 (Rev. 1) [1] related to defence in depth and the practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release. This Safety Guide also provides recommendations in relation to design aspects of defence in depth, in particular on those aspects associated with design extension conditions.		
17.	India	1	1.6	3	These measures play an important role in the implementation of the concept of defence in depth for achieving a balanced design of NPP, which constitutes the primary means of preventing accidents and mitigating their consequences should they occur, in accordance with Principle 8 of IAEA Safety Standards Series No. SF-1 Fundamental Safety Principles [3]	Optimal balance/ contribution is achieved by not burdening/ overwhelming a single system e.g., the containment during accidents, the other systems also should contribute in prevention and mitigation almost equally, ensuring a balance among DID Level and barriers.		Text deleted since this not the topic of the scope. However, presented in para 3.39. 1.7 … These measures play an important role in the application of the concept of defence in depth, which constitutes the primary means of both preventing and mitigating the consequences of accidents, in accordance with Principle 8 of IAEA Safety Standards Series No. SF-1, Fundamental Safety Principles [4].		
18.	UK	5	1.6 (& through out e.g. 1.14, 2.2, 3.3, 3.6)		Check for correct use of 'safety systems, 'safety features', 'safety measures', 'design measures' & 'safety provisions'.	Use correct/consistent terminology. Safety systems, safety measures and safety features have specific meaning (as per IAEA Glossary)	X			Corrected during technical edition
19.	UK	6	1.6		Change penultimate sentence to: "Safety measures and safety features correspond to one or more levels of defence in depth."	As written this states: "Safety features for DEC as well as other safety features that underpin the demonstration of practical elimination……" – this seems to be suggesting that there are two sorts of 'safety features', those for DEC and those for PE - it		1.8 As described in para. 2.13 of SSR-2/1 (Rev. 1) [1], defence in depth at nuclear power plants comprises five levels. Plant states considered in the design correspond to one or more levels of defence in depth. This Safety Guide is structured in		Corrected during technical edition

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						needs to recognise that any safety feature considered as part of DEC may be part of the PE demonstration. This also needs to apply to provisions across all levels of defence in depth (not just safety features).		terms of the design of safety provisions necessary for each plant state, rather than for each level of defence in depth. In this way, the significance and importance of design extension conditions for the safety approach is emphasized.		
20.	France	3	1.7		In addition to AOO and DBA, DEC without significant fuel degradation and DEC with core melting are part of the implementation of the concept of Defence in Depth. In terms of deterministic safety analyses methods, rules and assumptions to be followed, the IAEA safety guide SSG2 is already providing relevant guidance. However, there is a need to develop guidance about the integration of DEC within the overall implementation of Defence in Depth~~, as well as guidance on the identification of DEC conditions to be studied~~.	This guidance on identification of DEC is not necessary because it is still included in SSG-2 (deteleted sentence is contradictory with the previous one)				Original text deleted because is repeated from previous paragraphs and does not need to be introduced here.
21.	Canada	5	1.7	3rd sentence	*1.7 In addition to anticipated operational occurrence (AOOs), ~~and~~ design basis accidents (DBAs), design extension conditions (DECs) without significant fuel degradation and DEC with core melting are part of the implementation of the concept of Defence in Depth. In terms of deterministic safety analyses methods, rules and assumptions to be followed, the IAEA safety guide ~~SSG2 is al-ready~~ SSG-2 is already providing relevant guidance. However, there is a need to develop guidance about the integration of DEC within the overall implementation of Defence in Depth~~, as well as guidance on the identification of DEC conditions to be studied~~.*	SSG-2 already provides guidance for identification of DEC. Delete this part of the final sentence. The guidance provided in DS508 section 3 (mainly para 3.17) is very minor anyway and does not need to be highlighted here. Also, give full text of abbreviations in first use of the plant states and fix the typos in mid paragraph.				Original text deleted because is repeated from previous paragraphs and does not need to be introduced here.
22.	Germany	3	1.7		In addition to anticipated operational occurrence (AOO) and design basis accident (DBA), DEC without significant fuel degradation and DEC with core melting are part of the implementation of the concept of d~~D~~efence in d~~D~~epth. In terms of deterministic safety analyses methods, rules and assumptions to be followed, the IAEA safety guide ~~SSG2~~ SSG-2 (Rev. 1) [8] is al-ready providing relevant guidance. This Safety Guide was developed to ~~However, there is~~ fulfil a request ~~to develop~~ for guidance about the integration of DEC within the overall implementation of d~~D~~efence in d~~D~~epth, as well as guidance on the identification of DEC conditions ~~to be studied~~.	As AOO and DBA are not official acronyms (see IAEA Safety Glossary) it makes sense to use full terms first. Please formulate more clear the scope of this Safety Guide concerning current issue.		Acronyms were spelled out and formulation was made more clear. The scope of the SG was clarified.		

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23.	Italy	2	1.7	1		The acronym AOO is not defined	X			Considered during technical edition
24.	Italy	3	1.7	2	[…] concept of defence in depth […]	In the rest of the document, defence in depth is used without capital letters	X			Considered during technical edition
25.	Italy	4	1.7	4	[…] is already providing […]	Typo in "already"	X			Considered during technical edition
26.	Italy	5	1.7	5	[…] implementation of defence in depth […]	Same as comment no. 3	X			Considered during technical edition
27.	UK	7	1.7		Change "is al-ready providing" to "provides"	Minor typo	X			Considered during technical edition
28.	Indonesia	1	1.7	4	In addition to AOO and DBA, DEC without significant fuel degradation and DEC with core melting are part of the implementation of the concept of Defence in Depth. In terms of deterministic safety analyses methods, rules and assumptions to be followed, the IAEA safety guide ~~SSG2~~ SSG-2 (Rev. 1) is ~~al-ready~~ already providing relevant guidance. However, there is a need to develop guidance about the integration of DEC within the overall implementation of Defence in Depth, as well as guidance on the identification of DEC conditions to be studied.	SSG-2 has been replaced by SSG-2(Rev.1). replace the grammatically incorrect al-ready to already	X			
29.	India	31	1.7 & many other clauses		***Suggestion*** The terms 'significant fuel degradation', 'core melting', 'fuel damage', 'core damage' are used in this guide, sometimes interchangeably. A clear definition of terms and consistency in their usage is essential for correct interpretation of the guidance given in this document.	To avoid misinterpretation and ambiguity during application of the safety guide. The meaning of 'fuel' and 'core' are to be clear for a proper safety assessment. The term 'significant' adds further complexity to the understanding. For example, 'significant fuel degradation' includes fuel melting or not or it refers only to incipient melting or melting of a significant part and if so to what extent? As of now, though there is a common understanding, it is only qualitative. When practically applied to an NPP, interpretation could differ between a regulator and designer.				Considered by the technical editors in accordance with the IAEA Safety Glossary
30.	Italy	7	1.8	2	[…]in depth, with particular attention to safety […]	Sentence is not clear	X			Considered after technical edition
31.	France ENISS	4 3	1.8		Aspects such as environmental factors, operational or human factors, and external or internal hazards, are recognized as relevant, but are not addressed in detail in this Safety Guide.	For accuracy. They are referred to at various points but not addressed in detail. These factors are referred to later in the document as part of the assessment (e.g. para 4.23), and form, in some cases, a key part of the justification. Consider including appropriate IAEA references for methodologies at the appropriate points.	X	1.9 …However, factors that could cause dependence between structures, systems and components, such as environmental factors, operational or human factors, and external or internal hazards, are not addressed in detail in this Safety Guide.		
32.	Germany	4	1.8		A key requirement in prevention and mitigation of accidents is the independence, as far as practicable, between levels of defence in depth. In this Guide, focus is given ~~and in particular in~~	Please make clear a key requirement of what is meant here.				Original text deleted since it does not belong to the scope and the intention is

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					~~relation~~ to the independence of safety features for DEC (especially features for mitigating the consequences of accidents involving the melting of fuel). There are several factors that can be the cause of dependencies between plant structures, systems and components (SSCs) and that are addressed by different means. This Safety Guide considers, in a general manner, the assessment of functional independence of SSCs. Aspects such as environmental factors, operational or human factors, and external or internal hazards, are recognized as relevant, but are not addressed in this Safety Guide.	The main focus of DiD should be given to the independence of safety features for DEC from safety systems and other safety related items (thus DiD levels 1-3).				considered in appropriate paragraphs of sections 3 and 4.
33.	France ENISS	5 4	1.9		This Safety Guide also addresses the assessment of how ~~the demonstration of~~ defence in depth can be reinforced by including design features for enabling the use of non-permanent equipment for power supply and for cooling, as a result of the lessons learned from the Fukushima Daiichi accident	The demonstration is not reinforced, defence in depth is. The guide is loXing at how this is assessed.	X	The introduction of non-permanent equipment is not considered in this section. 1.3 The objective of this Safety Guide is to provide recommendations for the design of new nuclear power plants on the application of selected requirements in SSR-2/1 (Rev. 1) [1] related to defence in depth and the practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release. This Safety Guide also provides recommendations in relation to design aspects of defence in depth, in particular on those aspects associated with design extension conditions. 1.9 This Safety Guide considers the assessment of the independence of defence in depth and, in a general manner, the assessment of independence of structures, systems and components.		
34.	Canada	6	1.9		*1.9 This Safety Guide also addresses how the demonstration of defence in depth can be reinforced by including design features for enabling the use of non-permanent equipment for services such as power supply and for cooling~~, as a result of the lessons learned from the Fukushima Daiichi acciden~~t. These features …*	Power supply and cooling are just two (most important) examples. Others are possible (compressed air, lubrication). It is also not really necessary to include text about Fukushima. Many non-permanent services pre-date the Fukushima accident.				
35.	Canada	7	1.9		*These features are primarily intended for preventing unacceptable radioactive releases in ~~the~~ very rare and serious events, ~~of levels of~~ such as natural external hazards exceeding the magnitude considered for the design, derived from the hazard evaluation for the site.*	The provisions afforded by DECs should not be limited to cater to external hazards only. Also emphasis on the very rare aspects, since in theory the design should cover all hazards.				
36.	Germany	5	1.9		This Safety Guide also addresses the assessment of the design features to further strengthen ~~how the demonstration of~~ defence in depth ~~can be reinforced~~ by including design features for enabling the use of non-permanent equipment for power supply and for cooling, as a result of the lessons learned from the Fukushima Daiichi accident. These features are primarily intended for preventing unacceptable radioactive releases in the event ~~of levels~~ of natural external hazards exceeding the magnitude considered for the design, derived from the hazard evaluation for the site.	Clarification				

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37.	Italy	8	1.11	6	[…] This Safety Guide takes […]	Consistency with other capitalizations	X			Considered during technical edition
38.	France ENISS	6 5	1.12		The recommendations given in this Safety Guide are primarily intended for application to water cooled nuclear power plants designed in accordance with the requirements provided in SSR-2/1 (Rev. 1) [1]. It is recognized that for reactors cooled by other media or based on innovative design concepts, some of the recommendations in this Safety Guide might not be ~~fully~~ applicable, and judgment in their application ~~might~~ will be needed.	The wording '…might not be fully applicable...' suggests that all recommendations in this guide are applicable to at least some degree – this is not true when you are dealing with a different technology for a legacy plant.	X	1.5 This Safety Guide applies primarily to new land based stationary nuclear power plants with water cooled reactors, designed for electricity generation or for other heat production applications (such as district heating or desalination) (see para 1.6 of SSR-2/1 (Rev. 1) [1]). It is recognized that for reactors cooled by other media or reactors based on innovative design concepts, some of the recommendations in this Safety Guide might not be applicable or fully applicable, or judgement might be needed in their application. 1.6 For nuclear power plants designed in accordance with earlier standards, this Safety Guide might also be useful when evaluating potential safety enhancements of such designs, for example, as part of the periodic safety review of the plant.		
39.	Italy	9	1.12	3	It is recognized that for reactors cooled by other fluids or based on innovative design concepts, some of the recommendations in this Safety Guide might not be fully applicable, and judgment in their application might be needed.	"Fluids" may be considered instead of "media" in this context.	X	1.5 This Safety Guide applies primarily to new land based stationary nuclear power plants with water cooled reactors, …		Considered during technical edition
40.	Germany	6	1.13		The recommendations given in this Safety Guide are primarily intended for application to water cooled nuclear power plants designed in accordance with the requirements provided in SSR-2/1 (Rev. 1) [1]. It is recognized that for reactors cooled by other media or based on innovative design concepts, some of the recommendations in this Safety Guide might not be ~~fully~~ applicable, and judgment in their application ~~might~~ will be needed.	Clarification		This information was originally provided in 1.12. Modification provided to original para. 1.12 as follows: 1.5 This Safety Guide applies primarily to new land based stationary nuclear power plants with water cooled reactors, designed for electricity generation or for other heat production applications (such as district heating or desalination) (see para 1.6 of SSR-2/1 (Rev. 1) [1]). It is recognized that for reactors cooled by other media or reactors based on innovative design concepts, some of the recommendations in this Safety Guide might not be applicable or fully applicable, or judgement might be needed in their application.		
41.	UK	8	1.13		Add to paragraph: "Annexe II includes advice on how this guide should be applied to the assessment of existing plants designed to earlier standards."	For completeness	X			Clarification of the content of Annex II is provided in 1.13.
42.	ENISS	6	1.14		Section 4 provides recommendations on the application of the concept of practical elimination of event sequences that could lead to ~~early radioactive releases~~ high radiation doses or large radioactive releases.	To use the wording of SSR 2/1 Rev 1 Para 4.3 [*4.3. The design shall be such as to ensure that plant states that could lead to high radiation doses or to a large radioactive release have been 'practically eliminated', and that there would be no, or only*				The suggestion agreed was: Section 4 provides recommendations on the application of the concept of practical

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						minor, potential radiological consequences for plant states with a significant likelihood of occurrence.]. In relation to the wording in the following parts: It would be good to add the definitions of early radioactive releases and large radioactive releases in the document (definitions as in the 2018 IAEA Glossary). It is in paragraph 4.1, but it would be good to read it earlier in the document.				elimination of event sequences that could lead to early radioactive releases or large radioactive releases.
43.	France ENISS	7 7	1.15		Annex I provides information on the demonstration of a commonly recognized set of events or plant conditions that need to be demonstrated to have been practically eliminated. ~~Annex II provides some considerations for the application of this Safety Guide to nuclear power plants designed to earlier standards.~~ This guide focuses on the application of practical elimination in the design of new plants. Consideration of lessons to be learned from this process for existing plants should form a part of periodic reviews with the objective of further improving the level of safety, where reasonably practicable.	This was what was agreed at the NUSSC working group last year. If some member states are disassociating themselves with the agreement and want to include an Annex capturing preliminary thoughts, then an alternative wording could be "Annex II provides some preliminary consideration of this aspect which should be considered when revising the guidance on periodic safety review." If Annex II is included, it needs modification as indicated below.	X	The inclusion of Annex II was proposed by Austria. It was debated whether it should be an Appendix (as it is in SSG-53 for the containment) and it was agreed to develop an Annex. The proposal of Austria was supported by other countries. Recommendations are not provided in Annexes. Proposed text: 1.13 Annex I provides examples of cases of practical elimination. Annex II provides some considerations for the application of recommendations included in this Safety Guide to nuclear power plants designed to earlier standards.		
44.	France	8	2 (title)		DESIGN APPROACH WHEN CONSIDERING CONSEQUENCES OF ~~TO AVOID~~ ACCIDENTS ~~WITH HARMFUL CONSEQUENCES~~	In current standards, "harmful" is never used with "avoid" In the chapter, there only one occurrence of harmful and in a very fuzzy use Objective of 2 is clear in 1.14 This title is misleading If reading different articles, each says something when considering consequences of accident (or incident)	X	Harmful is repeatedly used in SF-1, for instance FSP-8 introducing DiD *3.30. The most harmful consequences arising from facilities and activities have come from the loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or other source of radiation. Consequently, to ensure that the likelihood of an accident having harmful consequences is extremely low, measures have to be taken:* Final title proposed as 2. DESIGN APPROACH CONSIDERING THE RADIOLOGICAL CONSEQUENCES OF ACCIDENTS		
45.	France	9	2.1		~~Principle 8 on prevention and mitigation of accidents in SF-1 [3] states that "All practical efforts must be made to prevent and mitigate nuclear or radiation accidents" and furthermore that "The primary means of preventing and mitigating the consequences of accidents is 'defence in depth'".~~	To be deleted : reference to SF-1 provides no added value in this guidance, is misleading and is not consistent with article 1.14	X	Text deleted		

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46.	France	10	2.2		~~The implementation of defence in depth, as described in SF-1 [3], comprises safety measures of various types. This Safety Guide is primarily focused on design measures for nuclear power plants as described in [1] and more specifically on design measures for the mitigation of accidents, including those implemented to facilitate accident management.~~	To be deleted : reference to SF-1 provides no added value in this guidance, is misleading and is not consistent with article 1.14	X	Text deleted		
47.	Canada	66	2.4		No change required.	It is good to bring all these fundamental technical requirements together. These are the key requirements related to frequency and consequences for accident conditions.	X			
48.	UK	9	2.4 & 4.1		Consider including a statement on if/how this guide addresses PE of 'high radiation doses'.	These paras are quoting paragraphs 2.11 & 4.3 of SSR-2/1: "The design shall be such as to ensure that plant states that could lead to <u>high radiation doses</u> or to a large radioactive release have been 'practically eliminated'…..". The 'high radiation dose' aspect isn't really considered further in the guide, e.g. in Section 4 the focus is on PE of early or large releases (as in 5.31 of SSR 2/1). For completeness the guide should explain the relevance of PE of 'high radiation doses'.			X	Terminology corrected after technical edition and quotation to requirements in SSR-2/1 (Rev.1) is made.
49.	France	11	2.6		The requirements in paras 2.3–2.5 ~~establish the safety approach for the design and specifically~~ establish the need for radiological consequences of accident conditions to be not only below acceptable limits but to be as low as reasonably achievable (ALARA). In addition, it needs to be demonstrated in the design that plant states that could lead to high radiation doses or to a large radioactive release have been 'practically eliminated'. Further requirements ~~in relation to acceptable limits~~ for categories of plant states and more specifically for accident conditions are also specified in SSR-2/1 (Rev. 1) [1], namely…	The previous requirements are not presented as establishing an approach in SSR-2/1.. Moreover, it is disputable if it is an objective or an approach or something else The requirements below do not mention "acceptable" limits	X	2.4 Furthermore, para. 4.4 of SSR-2/1 (Rev. 1) [1] states: "Acceptable limits for purposes of radiation protection associated with the relevant categories of plant states shall be established, consistent with the regulatory requirements." 2.5 Further requirements on criteria and objectives relating to radiological consequences of different plant states considered in the design, including accident conditions, are also established in SSR-2/1 (Rev. 1) [1], namely:		
50.	France	12	2.6		Adapt text to be in line with SSR-2/1 5.31: "…that could lead to an early radioactive release or a large radioactive release have been "practically eliminated" This is also what is written in the 2nd bullet in para 2.6.	In 2.6 it reads: "…it needs to be demonstrated in the design that plant states that could lead to <u>high radiation doses or to a large radioactive release</u> have been 'practically eliminated'." It is not in line with SSR-2/1 para. 5.31: "5.31. "The design shall be such that the possibility of conditions arising <u>that could lead to an early radioactive release or a large radioactive release</u> is 'practically eliminated'."	X	2.3 Paragraph 4.3 of SSR-2/1 (Rev. 1) [1] states: "The design shall be such as to ensure that plant states that could lead to high radiation doses or to a large radioactive release have been 'practically eliminated', and that there would be no, or only minor, potential radiological consequences for plant states with a significant likelihood of occurrence."		

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						2.11 and 4.3 are aligned to high radiation. While 5.31 is on LER.				
51.	Germany	7	2.6	Bullet 4 New issue	… — "The design shall be such that for design extension conditions, protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures" (para. 5.31A of SSR-2/1 (Rev. 1) [1] in relation to DEC). — "The same or similar technical and radiological criteria as those for design basis accidents may be considered for these conditions to the extent practicable. Radioactive releases should be minimized as far as reasonably achievable." (para. 7,46 of SSG-2 (Rev. 1) [8] in relation to DEC without core melt).	We propose to add a new item referring to the radiological acceptance criteria for DEC without core melt as stated in SSG-2 Rev.1. It is important to emphasize that in case of DEC without core melt same or similar radiological acceptance criteria apply as those for DBA. This information contributes to strengthening defense in depth and should be considered. This item also foster consistency with para 2.8.		Text added as part of para. 3.21 (c): 3.21 … (c) The acceptable criteria related to the radiological consequences for design extension conditions without significant fuel degradation may be identical or similar to those for design basis accidents (see paras 7.32 to 7.33 and 7.46 of SSG-2 (Rev. 1) [9]).		
52.	Canada	67	2.6	Editorial	Close quote at end of para.		X			
53.	France	13	2.7		This Safety Guide is focused on the protection of the public and the environment in accident conditions, which should be assessed notably regarding compliance with a number of requirements in SSR-2/1 (Rev. 1) [1] pertaining to the general plant design and particularly on its capability to withstand, ~~without unacceptable radiological consequences~~, accidents that are either more severe than design basis accidents or that involve additional failures, as well as other requirements for plant specific systems, for instance those related to the containment structure and its systems	SSR-2/1 is more ambitious than simply withstand DEC without unacceptable consequences. For example, as mentioned in 2.3 and 2.6, consequences shall be as low as reasonably practicable.	X	2.6 As indicated in para. 2.10 of SSR-2/1(Rev. 1) [1]: "…Measures are required to be taken to ensure that the radiological consequences of an accident would be mitigated. Such measures include the provision of safety features and safety systems, the establishment of accident management procedures by the operating organization and, possibly, the establishment of off-site protective actions by the appropriate authorities, supported as necessary by the operating organization, to mitigate exposures if an accident occurs."		
54.	Germany	8	2.7		This Safety Guide is focused on the protection of the public and the environment in accident conditions, which should be assessed notably regarding compliance with a number of requirements in SSR-2/1 (Rev. 1) [1]. These requirements pertain~~ing~~ to the general plant design and particularly on its capability to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures, as well as other requirements for plant specific systems, for instance those related to the containment structure and its systems.	Proposal for improved readability		2.1 This Safety Guide is focused on the design features in a nuclear power plant for the protection of the public and the environment in accident conditions, which should be assessed regarding compliance with a number of requirements in SSR-2/1 (Rev. 1) [1]. These requirements pertain to the general plant design and particularly on the capability of the plant to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures.		

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55.	France	14	2.8		In accordance with 2.13 ~~Requirement 5~~ of SSR-2/1 (Rev. 1) [1], ~~radioactive releases in accident conditions are required to be below acceptable limits and be as low as reasonably achievable. In addition,~~ "the safety objective in the case of a severe accident is that only protective actions that are limited in terms of lengths of time and areas of application would be necessary and that ~~the purpose of the fourth level of defence in depth~~ is that off-site contamination is avoided or minimized". ~~To this aim, a limit for the release of radioactive materials or on acceptable limit on effective dose should be specified for each category of accident conditions (acceptance criteria for deterministic safety analysis is addressed in section 4 of SSG-2[8]) and compliance with these limits should be verified. For accidents without significant fuel degradation the same or similar technical and radiological criteria as those for design basis accidents may be considered to the extent practicable. Radioactive releases should be minimized as far as reasonably achievable, such that off-site protective measures (e.g. sheltering, evacuation) are not necessary. For accident with core melting, the releases are required to be such that only protective actions that are limited in terms of lengths of time and areas of application would be necessary and that off-site contamination would be avoided or minimized. Event sequences that would lead to an early radioactive release or a large radioactive release are required to be 'practically eliminated'. The amount of radioactive releases considered acceptable for DEC with core melting should be significantly lower than the amount characterizing a large release. In addition, the design should be such that there are no cliff edge effects in radiological consequences for accidents slightly exceeding those considered in the design (including design extension conditions)~~	Req 5 has already been quoted in 2.3, there is neither guidance neither added value to rephrase it. Please be consistent with SSR-2/1 without rephrasing it Fully disagree with this part which is not consistent with SSG-2: the approach regarding acceptance criteria addressed in SSG-2 could not be rephrased with these straightforward sentences without challenging safety approach. Moreover, there is neither guidance neither added value by rephrasing already quoted requirement at the end of the article.		2.7 In accordance with para. 2.13 of SSR-2/1 (Rev. 1) [1]: "…The safety objective in the case of a severe accident is that only protective actions that are limited in terms of lengths of time and areas of application would be necessary and that off-site contamination is avoided or minimized". Text deleted. Requirement is quoted. 2.5 …— "The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated'" (para. 5.31 of SSR-2/1 (Rev. 1) [1] in relation to design extension conditions).		Considered during technical edition
56.	Italy	10	2.8	12	[…] limited in terms of time-spans and areas […]	Style	X			Considered during technical edition
57.	Canada	8	2.8 **Major comment**	Sentences 3, 4 and 5	~~For accidents without significant fuel degradation, the same or similar technical and radiological criteria as those for design basis accidents may be considered to the extent~~	The text is misleading. The requirement in SSR-2/1 is the same for DEC-A and DEC-B.				There is no need to present a footnote if the requirement is quoted

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					~~practicable. Radioactive releases should be minimized as far as reasonably achievable.~~ ~~such that off-site protective measures (e.g. sheltering, evacuation) are not necessary~~. ~~For accident with core melting,~~ For design extension conditions, *the releases are required to be such that only protective actions that are limited in terms of lengths of time and areas of application would be necessary and* ~~that off-site contamination would be avoided or minimized~~ *sufficient time would be available to take such measures. Event sequences that would lead to an early radioactive release or a large radioactive release are required to be 'practically eliminated'.*	*"5.31A. The design shall be such that for design extension conditions, protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public, and sufficient time shall be available to take such measures."* Member States may choose to apply DBA limits to DEC-A, but it is not required. DS508 must state the requirement. The optional position of some Member States could be included as a footnote, but then it would be necessary to point out that other Member States use the actual requirements of SSR-2/1.				
58.	UK	10	2.8		"For accidents with core melting, …."	Minor typo Note, later on in the same paragraph, "DEC with core melting" is mentioned. The 2018 Safety glossary entry for plant states does show DECs as an accident condition, so there is no logical inconsistency. However, it is not clear if there is any 'hidden meaning' for using inconsistent wording within the same paragraph.	X			Original text deleted
59.	Canada	9	2.8 **Major comment**	Sentence 7	The text in DS508 must be revised to agree with SSR-2/1 or SSR-2/1 must be revised to allow the DS508 interpretation. "~~The amount of radioactive releases considered acceptable for DEC with core melting should be significantly lower than the amount characterizing a large release.~~"	Canada strongly disagrees with this interpretation of SSR-2/1 paras 5.31 and 5.31A (also repeated twice in paragraph 4.7). In response to this comment on a previous draft, the authors responded that the "significant difference" represents a safety margin. It does not: it represents a gap. To provide a safety margin, consequences that must be practically eliminated should slightly overlap with the consequences permitted in DEC.	X	Text deleted		
60.	France	15	2.9		~~For normal operation or anticipated operational occurrences, there is limited uncertainty on plant state frequency and radiological impact, which can be monitored and is supported by many years of operating experience of previous plant designs. For less frequent plant states, i.e. accidents, there are larger uncertainties associated with the demonstration of plant state frequency and radiological consequences.~~	Please consider deletion of this article (not consistent with the scope of the document) or, at a very minimum, some complementary explanation: what does that mean that there is limited uncertainty of normal operation frequency? What does monitoring of radiological impact of a NPP when designing it? What is the consequence of stating that there are large uncertainties associated to accidents considering that a very basic principle is to address adequately uncertainties?...	X	Text deleted		
61.	ENISS	8	2.9		For normal operation or anticipated operational occurrences, there is limited uncertainty on plant state frequency and radiological impact, which can be monitored and is supported by	The initial text does not give confidence in the possibility to reduce risks, to demonstrate it.				Original text was deleted

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					many years of operating experience of previous plant designs. For less frequent plant states, i.e. accidents, there are larger uncertainties on ~~associated with the demonstration of~~ plant state frequency and radiological consequences, requiring adequate consideration in the demonstration.	Proposal of slight change is to keep the same text structure between first and second sentence				
62.	Germany	9	2.9		For normal operation or anticipated operational occurrences, there is limited uncertainty on plant state frequency and radiological impact, which can be monitored and is supported by many years of operating experience of previous plant designs. For less frequent plant states, i.e. accident conditions ~~accidents~~, there are larger uncertainties associated with the demonstration of plant state frequency and radiological consequences.	According to the IAEA Glossary, the term "accident conditions" comprises DBA and DEC.				Original text was deleted
63.	India	2	2.9	Line 5	For less frequent plant states, i.e. accidents, there are larger uncertainties associated with the demonstration of plant state frequency and radiological consequences. These uncertainties should be suitably factored in the assessments.	The uncertainty or the error in estimation can be mitigated with enhanced safety margins.				No doubt, but the intention is not to provide recommendations in this regard here. It was considered together with comments of other MSs and finally the original text was deleted
64.	France ENISS	16 9	2.10		Therefore, the following chapters are devoted to the implementation and assessment of design extension conditions within the concept of defence	Typo, a word is missing				Text deleted since it provides only introductory text.
65.	Germany	10	2.10		Harmful radiological consequences to the public can only arise from the occurrence of uncontrolled accident conditions ~~accidents~~. ~~Therefore, t~~The following chapters are devoted to the implementation and assessment of design extension conditions within the concept of defence in depth and the complementary need for demonstration of practical elimination of accident sequences that can lead to early radioactive releases or large radioactive releases	According to the IAEA Glossary, the term "accident conditions" comprises DBA and DEC.	X	2.8 Harmful radiological consequences to the public can arise only from the occurrence of uncontrolled accidents. Recommendations on radiation protection in the design of nuclear power plants are provided in IAEA Safety Standards Series No. NS-G-1.13, Radiation Protection Aspects of Design for Nuclear Power Plants [13], and recommendations for protection of the public and the environment are provided in IAEA Safety Standards Series No. GSG-8, Radiation Protection of the Public and the Environment [14].		
66.	UK	11	2.10		Change to "…assessment of design extension conditions…"	Minor typo	X			
67.	France	17	3.1		This section addresses the overall application of Requirement 7 of SSR-2/1 (Rev. 1) [1] for			Text deleted and a quotation of SSR-2/1 (Rev. 1) added		

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					defence in depth in the design of nuclear power plants, with specific emphasis on design provisions for accident conditions. It also addresses the overall assessment of the implementation of this concept, with specific focus on the reactor core as the main source of radioactivity. ~~For other sources of radiation or potential releases of radioactive materials, the implementation of a defence in depth strategy will depend on the amount and isotopic composition of radionuclides, on the effectiveness and leak tightness of the individual confinement barriers as well as the potential challenges for the integrity of the barriers and the consequences of their failures~~ Alternative text could be : As per 2.14 of SSR-2/1 "*A relevant aspect of the implementation of defence in depth for a nuclear power plant is the provision in the design of a series of physical barriers.[…]. The number of barriers that will be necessary will depend upon the initial source term in terms of the amount and isotopic composition of radionuclides, the effectiveness of the individual barriers, the possible internal and external hazards, and the potential consequences of failures.*"	This sentence is not consistent with SSR-2/1-art.2.14: in this article, it is said that the number of barriers will depend, this article does not say not the implementation of DiD		3.1 The concept of defence in depth for the design of nuclear power plants is described in paras 2.12-2.14 of SSR-2/1 (Rev. 1) [1]. As stated in para. 2.14 of SSR-2/1(Rev. 1) [1]: "A relevant aspect of the implementation of defence in depth for a nuclear power plant is …and the potential consequences of failures."		
68.	Germany	11	3.1		This section addresses the overall application of Requirement 7 of SSR-2/1 (Rev. 1) [1] for defence in depth in the design of nuclear power plants, with specific emphasis on design provisions for design extension ~~accident~~ conditions. It also addresses the overall assessment of the implementation of this concept, with specific focus on the reactor core as the main source of radioactivity. For other sources of radiation or potential releases of radioactive materials, the implementation of a defence in depth strategy will depend on the amount and isotopic composition of radionuclides, on the effectiveness and leak tightness of the individual confinement ~~barriers~~ structures as well as the potential challenges for the integrity of the physical barriers and the consequences of their failures.	According to the IAEA Glossary, "accident conditions" comprise DBA and DEC. According to the title of section 3 implementation of DEC into DiD shall be addressed in this section. Further clarification in accordance with SSR-2/1 (Rev. 1) and IAEA Safey Glossary		Original text deleted and a quotation of SSR-2/1 (Rev. 1) added 3.1 The concept of defence in depth for the design of nuclear power plants is described in paras 2.12-2.14 of SSR-2/1 (Rev. 1) [1]. As stated in para. 2.14 of SSR-2/1(Rev. 1) [1]: "A relevant aspect of the implementation of defence in depth for a nuclear power plant is …and the potential consequences of failures."		

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69.	Germany	12	3.1	3rd sentence	… For other sources of radiation or potential releases of radioactive materials, as for instance the spent fuel pool, the implementation of a defence in depth strategy will depend on the amount and isotopic composition of radionuclides, on the effectiveness and leak tightness of the individual confinement barriers as well as the potential challenges for the integrity of the barriers and the consequences of their failures.	To emphasize the spent fuel pool as a main source of radioactive inventory in addition to the core.		Original text deleted		
70.	UK	12	3.1		Suggest the wording of SSR 2/1 Requirement 7 is included	For completeness and for consistency with other parts of the document where the full wording is included for other Requirements.	X			
71.	France	18	3.2		The concept of defence in depth for the design of nuclear power plants is described in para. 2.12-2.14 of SSR-2/1 (Rev. 1) [1]. ~~An overall strategy of defence in depth, when properly implemented, achieves the objective that no single technical, human or organizational failure will lead to harm to the public, and that credible combinations of events and failures will lead to no or little harm to the public~~	It is a non complete re-phrasing of SF-1 (thus non correct, thus introduces potential misleading) – 3.31 and objective of DiD which is out of scope of this guidance should not be oversimplified	X	Text deleted. Quotation considered in 3.1.		
72.	UK	13	3.2		Include 4.9-4.13A when referencing out to SSR 2/1.	These parts of SSR 2/1 (under Requirement 7) set the scene for much of what follows in Section 3.	X			
73.	UK	14	3.2		Add to the end of this para: "The design should provide for multiple physical barriers to the release of radioactive and ensure that the safety measures/features are effective in protecting these barriers."	To reflect the expectations of 4.11 & 4.12 of SSR 2/1 on physical barriers and to make the distinction between physical barriers and safety measures/features (but still make the link between the two).			X	Text deleted based on comment from France
74.	France ENISS	19 10	3.3		For the implementation of safety provisions at each level of defence in depth there are three aspects of importance, as follows: (a) The performance of the safety provisions implemented at ~~to meet the objective of each~~ that level, to protect ~~notably regarding the acceptance criteria for~~ the integrity of the barrier(s) that should be protected; (b) The reliability of the protection provided at that level. ~~safety measures to demonstrate with a sufficient level of confidence that a certain plant condition can be brought under control~~	To improve clarity. A systematic process will involve each level being considered in turn so need only consider independence from previous levels since when considering the provisions for the next level the process will be repeated and so loX at whether claims on and potential failure of the same equipment has been made. There are a number of acceptance criteria which are not discussed such as the overall adequacy of protection and how the individual levels contribute to this in terms of both performance and reliability. In practice this can only be done by an analysis of the complete system.		Text modified as: 3.4 For the safety provisions at each level of defence in depth, the following should be demonstrated: (a) The performance of the safety provisions implemented at that level to maintain the integrity of the barrier(s); (b) Adequate reliability of the safety provisions at that level so that it can be assured, with a sufficient level of confidence, that a certain plant condition can be brought under control without the need to implement safety provisions associated with the next level;		

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					~~without needing the intervention of the safety provisions implemented for next level~~;			(c) The independence, as far as applicable, of the safety provisions at that level, including their physical separation , from the safety provisions associated with the previous levels of defence in depth.		
75.	India	3	3.3		*Suggestion* 'Demonstration of absence of cliff-edge effects' may be included in this section	SSR/ 2.1 requirement 7: application of DID specifically requires avoidance of cliff-edge effects				Important topic, however, the intention of this safety guide is not to develop them but to provide recommendations for the implementation of the concept of practical elimination.
76.	UK	15	3.3 (b)		Two alternatives are suggested: "The reliability of safety provisions to ensure that a certain plant condition can be brought under control without needing the intervention of the safety provisions implemented for next level, **should be demonstrated** with a sufficient level of confidence" or: ~~The~~ "**Adequate [or maybe 'Sufficient']** reliability of safety provisions to ensure that a certain plant condition can be brought under control without needing the intervention of the safety provisions implemented for next level, with a sufficient level of confidence."	A repeat of comment made by the UK at Step 7 prior to MS comment stage: The current wording on 3.3(b) is not clear, in its own right, and in the context of the preceding text at the start to list. "..there are three aspects of importance, as follows: ….. The reliability of safety measures to demonstrate with a sufficient level of confidence that a certain plant condition can be brought under control without needing the intervention of the safety provisions implemented for next level" What is of importance? The reliability of the safety measures, the demonstration of reliability, or the level of confidence.		Text modified as: 3.4 For the safety provisions at each level of defence in depth, the following should be demonstrated: (a) The performance of the safety provisions implemented at that level to maintain the integrity of the barrier(s); (b) Adequate reliability of the safety provisions at that level so that it can be assured, with a sufficient level of confidence, that a certain plant condition can be brought under control without the need to implement safety provisions associated with the next level; (c) The independence, as far as applicable, of the safety provisions at that level, including their physical separation , from the safety provisions associated with the previous levels of defence in depth.		
77.	Germany	13	3.3	(c)	~~Adequate~~ independence, <u>as far as practicable,</u> from the safety provisions implemented at the previous and successive levels of defence in depth.	To be consistent with para. 1.8 "A key requirement is the independence, as far as practicable, between levels of defence in depth ..."		3.4 …(c) The independence, as far as applicable, of the safety provisions at that level, including their physical separation , from the safety provisions associated with the previous levels of defence in depth.		
78.	India	4	3.3	(c)	Adequate independence from the safety provisions implemented at the previous and successive levels of defence in depth <u>.In case safety provisions being implemented across DiD levels, it should be demonstrated that they satisfy the safety requirements commensurate with corresponding DiD levels independently'.</u>	To bring better clarity. This guide mentions about "adequate independence from the safety provisions implemented at the previous and successive levels of defence in depth" as an important aspect in the implementation of safety provision at each level of DiD. The term 'adequate' is qualitative. Though independence of safety provisions is desired, practically it may not be feasible in the design.		Adequate is indeed qualitative, but very rarely we provide quantitative statements. Adequate is extensively used in SSR 2/1. The subject is only introduced here. It is discussed later on in section 3, paras 3.56 to 3.66. Final text as: 3.4 …(c) The independence, as far as applicable, of the safety provisions at that level, including their physical separation , from the safety		Proposed text is not considered here since this topic is developed in section 3, paras 3.56 to 3.66..

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						Hence, the term 'adequacy' can be supplemented with an additional clause.		provisions associated with the previous levels of defence in depth.		
79.	UK	16	3.3 (c)		Change to read: "Adequate independence (including separation and segregation where possible) from the safety provisions…."	For completeness/clarity to provide a clear definition for independence, both in a physical and system architectural perspective		3.4 …(c) The independence, as far as applicable, of the safety provisions at that level, including their physical separation , from the safety provisions associated with the previous levels of defence in depth.		
80.	ENISS	11	3.4/ 3.19		Harmonize the text 3.4: … Radiological acceptable limits for DEC without core melt may be ~~are~~ the same or similar as ~~for~~ DBA limits… 3.19 : …the primary difference between these two accidental conditions is the possibility to use ~~of~~ different acceptance criteria, different design requirements or different approaches for performing safety analysis for this objective… (b) Less conservative assumptions and criteria than for DBA…	The 2 sections of 3.4 and 3.19 may be read as being inconsistent: "same" limits on one side, "different" criteria on the other side. Radiological limits are only some of the acceptance criteria. May be worth to be more flexible/precise in the text. If the idea is to say that most of the acceptance criteria may be kept for DEC-A, it could be worth distinguishing the items in 3.19: (b) Less conservative assumptions ~~and criteria~~ than for DBA, or best estimate methods, are acceptable for the safety analysis. (c) Identical or similar radiological limits as for DBA, whereas acceptance criteria may be similar or less conservative.	X	Comments implemented.		
81.	ENISS	14	3.4		An association of the levels of defence in depth with plant states considered in the design is frequently undertaken for design safety and operational safety. The introduction of DEC in the plant design basis has resulted in ~~two~~ different interpretations by States regarding the correspondence between plant states considered in the design and levels of defence in depth. ~~These~~ ~~t~~ Two approaches are represented in Table 1.	The regulation around the world have derived different interpretations, not only 2 (In Europe there are several derivations; UK, France, Belgium, Finland… have not exactly the same levels). For instance, the UK does not traditionally use DEC-A as we include such faults in the design basis using a graded approach to the analysis. The suggestion is to recognise that and emphasize the main 2 ones to keep the proposed text.		3.5 Frequently, for purposes of design safety and operational safety, the various levels of defence in depth are associated to the various plant states considered in the design. The introduction of design extension conditions among the plant states has resulted into different interpretations in different States regarding the correspondence between the plant states considered in the design and the levels of defence in depth. Two of these approaches are represented in Table 1.		
82.	Germany	14	3.4	2nd sentence	… The introduction of DEC in the plant design ~~basis~~ has resulted in two different interpretations by States regarding the correspondence between plant states considered in the design and levels of defence in depth.	With respect to the different points of view concerning the correspondence between plant states and levels of defence in depth as well as the meaning and understanding of the term *design basis*, here using only the expression *plant design* is sufficient.				
83.	Indonesia	2	3.4	5-6	An association of the levels of defence in depth with plant states considered in the design is frequently undertaken for design safety and operational safety. The introduction of DEC in the plant design basis has resulted in two different interpretations by States regarding the correspondence between plant states considered in the design and levels of defence in depth. These two approaches are represented in Table	The addition of the highlighted sentence after the last sentence in Para 3.4 should be considered.		3.7 Despite their differences, both of these approaches are in compliance with para. 5.29 (a) of SSR-2/1(Rev. 1) [1] and support, the implementation, to the extent practicable, of independence among safety systems, safety features for prevention of and safety features for mitigation of events considered in the design extension conditions.		

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					1. Approach 1 (i.e., the association of DEC without core melt to level 3) has the advantage that each level has clear objectives regarding the progression of the accident and the protection of the barriers, i.e., level 3 to prevent damage to the reactor core and level 4 to mitigate severe accidents for preventing off site contamination. Radiological acceptable limits for DEC without core melt are the same or similar as for DBA. Also, the physical phenomena in case of DBA and DEC without significant fuel degradation are similar, although there are differences in the analysis. In contrast, severe accidents are characterized by completely different physical phenomena. However, approach 2 (i.e., the grouping of DEC without core melt and with core melt in level 4) emphasizes the differentiation between the set of rules for design and for safety assessment to be applied for DEC and the rules to be applied to DBA. Hence, both approaches are clear. Each could have its own advantage. Hence, it may be considered to state that both approaches are equally valid, as long as they could be implemented consistently.					
84.	Germany	15	3.4	Line 11	… Also, the physical phenomena in case of DBA and DEC without significant fuel degradation are similar, although there may be ~~are~~ differences in the analysis.	It is not a matter of fact that there are differences. This is up to the national practice.		3.5 … Furthermore, the physical phenomena associated with design basis accidents and design extension conditions without significant fuel degradation are similar, although there might be differences in the analysis. In contrast, the physical phenomena associated with design extension conditions with core melt are completely different.		
85.	France	20	3.4		An association of the levels of defence in depth with plant states considered in the design is frequently undertaken for design safety and operational safety. The introduction of DEC in the plant design basis has resulted in ~~two~~ different interpretations by States regarding the correspondence between plant states considered in the design and levels of defence in depth. ~~These~~ ~~t~~Two approaches are represented in Table 1. If adequately applied, both of them lead to implementation of relevant provisions at each level of DiD. Approach 1 (i.e. the association of DEC without core melt to level 3) ~~has the advantage~~ is established considering that each level has clear objectives regarding the	The regulation around the world have derived different interpretations, not only 2 (In Europe there are several derivations UK, France, Belgium, Finland… have not exactly the same levels). The suggestion is to recognise that and emphasize the main 2 ones to keep the proposed text. There are more than two ap-proaches. For instance, the UK does not traditionally use DEC-A as we include such faults in the design basis using a graded approach to the analysis.It is of high importance to avoid enhancing opposition between approaches and to enhance that implementation of adequate provisions at each level of DiD is what is important for safety at the end.		3.5 Frequently, for purposes of design safety and operational safety, the various levels of defence in depth are associated to the various plant states considered in the design. The introduction of design extension conditions among the plant states has resulted into different interpretations in different States regarding the correspondence between the plant states considered in the design and the levels of defence in depth. Two of these approaches are represented in Table 1.		

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					progression of the accident and the protection of the barriers, i.e. level 3 to prevent damage to the reactor core and level 4 to mitigate severe accidents ~~for preventing off site contamination~~. Radiological acceptable limits for DEC without core melt are the same or similar as for DBA. Also, the physical phenomena in case of DBA and DEC without significant fuel degradation are similar, although there are differences in the analysis. In contrast, severe accidents are characterized by completely different objectives and physical phenomena. ~~However~~, approach 2 (i.e. the grouping of DEC without core melt and with core melt in level 4) emphasizes the differentiation between the set of rules for design and for safety assessment to be applied for DEC and the rules to be applied to DBA					
86.	France ENISS	21 13	3.4		Approach 1 (i.e. the association of DEC without core melt to level 3) has the advantage that each level has clear objectives regarding the progression of the accident and the protection of the barriers, i.e. level 3 to prevent fuel damage ~~to the reactor core~~ and level 4 to mitigate severe accidents for preventing off site contamination	The formulation is aligned with SSR-2/1 2.11 but deviates from other formulations such as DS508 3.19. Suggest aligning to 3.19 to expand the scope. It should rather be "fuel damage" or "core damage or damage to the fuel in the irradiated fuel storage" in accordance with 3.19. Just to include fuel damage in the fuel storage pool		3.5 … In Approach 1, depicted on the left hand side of Table 1, design extension conditions without significant fuel degradation are associated to level 3 of defence in depth. In this approach, each level has a clear objective that reflects the progression of an accident and the protection of the barriers, i.e. level 3 is implemented to prevent fuel damage and level 4 is implemented to mitigate severe accidents and prevent off-site contamination. Design extension conditions without significant fuel degradation could be understood as those representative event sequences involving either a single initiating event of very low frequency, or an anticipated operational occurrence or frequent design basis accident combined with multiple failures, which are considered in the design in order to prevent reactor core melt and melting of fuel stored in the spent fuel pool. Therefore, in Approach 1, acceptable limits on predicted radiological consequences for design extension conditions without significant fuel degradation may be the same as or similar to acceptable limits for design basis accidents. Furthermore, the physical phenomena associated with design basis accidents and design extension conditions without significant fuel degradation are similar, although there might be differences in the analysis. In contrast, the physical phenomena associated with design extension conditions with core melt are completely different.		

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87.	France ENISS	22 12	3.4		Last sentence : However, approach 2 (i.e. the grouping of DEC without core melt and with core melt in level 4) emphasizes the differentiation between the set of rules ~~for design and for safety assessment to be applied~~ for DEC and the rules to be applied to DBA.	Simplification to clarify the meaning. The presence of two "and" makes it difficult to understand what is A and B in the "between X and Y and W". If the text on design have to be kept, better to add it under parenthesis.		3.6 In Approach 2, depicted on the right hand side of Table 1, design extension conditions without significant fuel degradation and design extension conditions with core melt are grouped together in level 4 of defence in depth. This approach emphasizes the distinction between the set of rules to be applied for design extension conditions and the set of rules to be applied for design basis accidents, both in the design and in the safety assessment.		
88.	Canada	10	3.4	Final sentence	*However, approach 2 (i.e. the grouping of DEC without core melt and with core melt in level 4) emphasizes the differentiation between the set of rules for design and for safety assessment to be applied for DEC and the rules to be applied to DBA. Approach 2 also supports the requirement for independence (to the extent practicable) between systems associated with level 3 (e.g. safety systems) and level 4 (e.g. safety features for DEC).*"	Approach 2 also supports the requirement of SSR-2/1, para 5.29, item 9a) for independence (to the extent practicable) between levels of defence in depth. *5.29. The analysis undertaken shall include identification of the features that are designed for use in, or that are capable of preventing or mitigating, events considered in the design extension conditions. These features:* *(a) Shall be independent, to the extent practicable, of those used in more frequent accidents;* *(b) …*		3.7 Despite their differences, both of these approaches are in compliance with para. 5.29 (a) of SSR-2/1(Rev. 1) [1] and support, the implementation, to the extent practicable, of independence among safety systems, safety features for prevention of and safety features for mitigation of events considered in the design extension conditions.		
89.	Canada	68	3.4		Suggest adding a new sentence: "*Approach 2 also supports the SSR-2/1 para 5.29 (a) requirement for independence (to the extent practicable) between safety features for DECs and systems for AOO and DBA.*"	SSR-2/1 para 5.29 states: *5.29. The analysis undertaken shall include identification of the features that are designed for use in, or that are capable[15] of preventing or mitigating, events considered in the design extension conditions. These features:* *(a) Shall be independent, to the extent practicable, of those used in more frequent accidents;* *(b) ….* Approach 2 puts DEC-A and DEC-B together in level 4 DiD. This supports the SSR-2/1 requirement for independence between safety features for DEC and mitigating systems for DBAs.				
90.	India	5	Table 1	Level 1 /4th column (column of 'Essenti	~~Operational rules~~ Operational Limits and Conditions or the plant Technical Specifications and normal operating procedures	The document uses the terminology - Operational Limits and Conditions or the plant Technical Specification in Clause 3.5. Use of same terminology, in the table for DID level-1 will be better for consistency.	X	Text modified as: Operational limits and conditions and normal operating procedures		

No	MS/ Org.	Com ment No.	Para	Line No.	Proposed new text	Reason	Accept ed	Accepted, but modified as follows	Rejec ted	Reason for modification/rejection
				al operational means')						
91.	Canada	11	Table 1,	Level 3 3b Essential design means	*Safety systems and/or safety features*	Safety systems (e.g., reactor shutdown system) are also credited to deal with DEC without core melt (e.g., beyond design basis earthquake)		Footnote 7 added: Such safety features are understood as additional safety features for design extension conditions, or as safety systems with an extended capability to prevent severe accidents (see para. 5.27 of SSR-2/1 (Rev. 1)) [1].		
92.	Canada	12	Table 1,	Level 4 Essential design means	~~*Technical Support Centre*~~	Technical support centre is not dedicated for DEC with core melt. It is also invXed following a DBA			Y	The technical support centre is not an essential mean for DBA. DBAs should be possible to be managed by the MCR staff. It can be functional after sufficient time following a DBA and could be activated in the preventive domain. Its main purpose is for the mitigative domain.
93.	Japan	1	After Table1, before 3.5		*Insert subtitle* **"Normal operation and Anticipated operational occurrences"**	User-friendliness.	X			
94.	Germany	16	3.5		Normal operation comprises a series of plant operating modes defined in the documentation governing the operation of the plant (such as the Operational Limits and Conditions or the plant Technical Specifications in some States) that range from power operation to reactor refuelling, in which no failures with impact on the plant operation or the availability of safety features have taken place, and no equipment is unavailable that would prevent the intended accomplishment of the goals of the operational mode.	Clarification, as not all failures in subordinate systems will be treated as level 2.		3.8 Operational states comprise two sets of plant states: normal operation and anticipated operational occurrences. Modes of normal operation include startup, power operation, shutting down, shutdown, maintenance, testing and refuelling and are defined in the documentation governing the operation of the plant (e.g. the operational limits and conditions). Plant states other than normal operation are reached either directly by the occurrence of a postulated initiating event or through a failure in mitigating the consequences of such an event.		
95.	Egypt	1	3.5		Plant states other than normal operations are (Or) normal operation is reached either directly by the occurrence of postulated initiating events for the applicable modes of operation or through	Grammar	X			

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					failures in mitigating the consequences of such events in the first place.					
96.	Canada	13	3.5	10th line	*… this Safety Guide ~~addresses~~ is oriented by the design of safety provisions necessary for each plant state, rather than for each level of defence.*	Improve the clarity		Moved to the scope and modified as: 1.8 …This Safety Guide is structured in terms of the design of safety provisions necessary for each plant state, rather than for each level of defence in depth. In this way, the significance and importance of design extension conditions for the safety approach is emphasized.		
97.	France	23	3.5		Delete 3.5 and replace with: Operational states or operating conditions for NPPs are subdivided into two sets of plant states: normal operation and anticipated operational occurrences.	Plant states are defined in the IAEA Safety Glossary. This paragraph is confusing. Table 1 is meant to indicate the correspondence between plant states and levels of defence in depth. Some of that is included in the Objective column but what IAEA call Operational States includes "Normal Operation" and "Anticipated Operational Transients" are not mentioned. The latter is covered by Level 2, but the Objective of Level 1 as discussed in SSR 2/1 is far more wide ranging. The simplified paragraph introduces these two plant states and provides an introduction from which 3.6 and 3.7 flow. {		3.8 Operational states comprise two sets of plant states: normal operation and anticipated operational occurrences….		
98.	ENISS	15	3.5		Operational states or operating conditions for NPPs are subdivided into two sets of plant states: normal operation and anticipated operational occurrences. Normal operation comprises a series of plant operating modes defined in the documentation governing the operation of the plant (such as the Operational Limits and Conditions or the plant Technical Specifications in some States) that range from power operation to reactor refuelling.~~, in which no failures have taken place, and no equipment is unavailable that would prevent the intended accomplishment of the goals of the operational mode.~~ Plant states other than normal operation are reached either directly by the occurrence of postulated initiating events for the applicable modes of operation or through failures in mitigating the consequences of such events in the first place. Their impact on the plant is the main basis for establishing the safety provisions that are necessary at each plant state. For these reasons, this Safety Guide addresses the design safety provisions necessary for each plant state, rather than for each level of defence. ~~In this way, the significance and importance of~~	Plant states are defined in the IAEA Safety Glossary. This paragraph is confusing. Table 1 is meant to indicate the correspondence between plant states and levels of defence in depth. Some of that is included in the Objective column but what IAEA call Operational States includes "Normal Operation" and "Anticipated Operational Transients". The latter is covered by Level 2, but the Objective of Level 1 as discussed in SSR 2/1 is far more wide-ranging. This should be mentioned. Note that 'operational mode" is not defined and may be confusing in regard to "operational states". The proposed simplified paragraph introduces these two plant states and provides an introduction from which 3.6 and 3.7 flow. Generally, the main objective of the operational mode is to producing electricity without impairing the safety functions. Failures of components that are redundant or not essential may have occurred in normal operation, making the initial sentence (no failure) not exactly true.		The proposed text for deleting was accepted.		The proposal for deleting the last sentence is rejected since it is in line with the objective of the SG, to emphasize the

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					~~design extension conditions for the safety approach is emphasized.~~	Some design even incorporates redundant components to secure the normal operation or for maintenance. This does mean that a failure/unavailability may occur, even on an item important to safety, with a component requiring repair/maintenance, while the safe normal operation is not affected or only partially affected. Example1: maintenance of emergency diesel in normal operation (at-power) at SZB, thanks to sufficient redundancy. Example2: reduced turbine cooling (clogging) with part of the cooling system isolated and under repair (Plant at-power but reduced power). Consider deletion or an alternative text such as: "in which potential failures and equipment unavailability do not totally prevent the intended accomplishment of the goals of the operational mode."				DEC for the safety approach. The last two sentences were moved to the scope para 1.8.
99.	UK	17	3.5		Change to: "….(such as the Operational Limits and Conditions or the plant Technical Specifications in some Member States)" or: "….(such as the Operational Limits and Conditions or the plant Technical Specifications)"	'States' is used for Member States and plant states in para 3.5. To avoid any confusion, suggest Member States is written out in full (or just deleted).	X			Considered during the technical edition
100.	France	24	3.6		~~para 4.13 of SSR-2/1 (Rev.1) states:"The design shall be such as to ensure, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the nuclear power plant.". Therefore, design provisions for operational states should have adequate capabilities to keep integrity of the first barrier for confinement of radioactive materials (i.e. the fuel cladding) and to prevent a significant release of primary coolant and an evolution to design basis accident conditions, for which the actuation of the engineered safety features (safety systems) is foreseen~~	The first part of the article is a quotation then an explanation whether: -no link with the quotation that does not mention releases -or rephrasing with wording which does not seem to be adequate (the word "avoid" cannot simply replace prevent This article does not provide any guidance.		3.9 Paragraph 4.13 of SSR-2/1 (Rev. 1) [1] states: "The design shall be such as to ensure, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the nuclear power plant." Therefore, to maintain the integrity of the first physical barrier for the confinement of radioactive material (i.e. the fuel cladding) and to prevent a significant release of primary coolant, design provisions for operational states should have adequate capabilities to: (a) Prevent failures or deviations from normal operation by means of robust design and in compliance with proven engineering practices and high quality standards commensurate with the importance of these design provisions to safety;		
101.	Italy	11	3.6	7	[…] materials (i.e. the […]	Typo (space between parenthesis and "i.e.")	X			Corrected
102.	Canada	14	3.6	2nd para	*Therefore, design provisions for operational states ~~should have adequate capabilities to maintain the integrity of the first barrier for~~*	To be consistent with Table 1 Improve the clarity				

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					~~*the confinement of radioactive materials (i.e. the fuel cladding) and to prevent a significant release of primary coolant and an evolution to design basis accident conditions, for which the actuation of the engineered safety features (safety systems) is foreseen.*~~ *is to,* 1. *prevent failures or deviation from normal operation by conservative design and high quality standards commensurate with their importance to the safety* 2. *Detect and intercept deviations from normal operation and return the plant to a state of normal operation* *arrest the progression of plant transients (i.e., AOO) once they start*			(b) Detect and intercept deviations from normal operation and return the plant to a state of normal operation; (c) Prevent anticipated operational occurrences, once they start, from evolving into design basis accidents.		
103.	ENISS	16	3.6		Therefore, design provisions for operational states should have adequate capabilities to maintain the integrity of the first barrier for the confinement of radioactive materials (i.e. the fuel cladding) and to prevent a significant release of primary coolant and an evolution to ~~design basis~~ accident conditions, for which the actuation of the engineered safety features, ~~(~~safety systems~~)~~ and the application of the emergency procedures is foreseen.	To align with SSR-2/1 2.13 "*This leads to the requirement that inherent and/or engineered safety features, safety systems and procedures be capable of preventing damage to the reactor core or preventing radioactive releases requiring off-site protective actions and returning the plant to a safe state.*"				
104.	Germany	17	3.6	Line 6	Therefore, design provisions for operational states should have adequate capabilities to maintain the integrity of the first physical barrier for the confinement of radioactive materials (i.e. the fuel cladding) and to prevent a significant release of primary coolant and an evolution to design basis accident conditions, for which the actuation of the engineered safety features (safety systems) is foreseen.	Clarification		3.9 … Therefore, to maintain the integrity of the first physical barrier for the confinement of radioactive material (i.e. the fuel cladding) and to prevent a significant release of primary coolant, design provisions for operational states should have adequate capabilities to:		
105.	France	25	3.7		The provisions for normal operation and AOO should have a reliability commensurate with ~~consistent with the~~ highest frequency of postulated initiating events for design basis accidents, ~~(usually lower than 10-2 per reactor-year), the reliability of safety provisions for anticipated operational occurences should be such that the frequency of transition into an accident condition is significantly lower than this value~~.	Estimated frequency of accidents does not rely only on AOO provisions reliability Thus a more general recommendation is more adequate "Usually lower than 10-2" seems to be a not very ambitious expectation regarding frequency of postulated accidents		3.10 The reliability of safety provisions for anticipated operational occurrences should be such that the frequency of transition to a design basis accident is lower than the highest frequency of postulated initiating events for design basis accidents (usually lower than 10-2 per reactor-year).		
106.	Italy	12	3.7	1	Consistently with the […]	Syntax (adjective instead of adverb)	X			Corrected during technical edition

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107.	Canada	15	3.7		Delete paragraph 3.7. If deletion is rejected, a source for the requirement should be provided. Also, the paragraph should be reworded for clarity to reference the lowest frequency of AOO, not the highest frequency of DBA. Yes, they have the same value, but the text is about AOO, not DBA.	Why "significantly" lower? Surely if a PIE demands action from a control system then the assessed frequency of the [PIE + failure of control system] will fall into the plant state demanded by the combined frequency. The design must meet the acceptance criteria for that plant state. There is no justification for the proposed distortion of normal analysis rules. There is nothing in SSG-2 that requires this.				
108.	ENISS	17	3.7		Consistent with the highest frequency of postulated initiating events for design basis accidents (usually lower than 10-2 per reactor-year), the reliability of safety provisions for anticipated operational occurrences should be such that the frequency of transition into ~~an~~ a design basis accident ~~condition~~ is significantly lower than this value.	Just to use to the same vocabulary at the beginning/end of the sentence. Otherwise, this may be seen as a transition into a "severe accident", what is not the intent here.	X	3.10 The reliability of safety provisions for anticipated operational occurrences should be such that the frequency of transition to a design basis accident is lower than the highest frequency of postulated initiating events for design basis accidents (usually lower than 10-2 per reactor-year).		
109.	Germany	18	3.7		~~Consistent with the highest frequency of postulated initiating events for design basis accidents (usually lower than 10-2 per reactor-year), t~~The reliability of safety provisions for anticipated operational occurrences should be such that the frequency of transition into an accident condition is significantly lower than the highest frequency of postulated initiating events for design basis accidents (usually lower than 10^{-2} per reactor-year).	Proposal for improved readability		Initial text of paragraph was deleted and para text proposed modified as: 3.10 The reliability of safety provisions for anticipated operational occurrences should be such that the frequency of transition to a design basis accident is lower than the highest frequency of postulated initiating events for design basis accidents (usually lower than 10-2 per reactor-year).		
110.	France ENISS	26 18	3.10		Paragraph 5.25 of SSR-2/1 (Rev. 1) [1] states: "*The design shall be such that for design basis accident conditions, key plant parameters do not exceed the specified design limits. A primary objective shall be to manage all design basis accidents so that they have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site protective actions.*" Consequently, specific design provisions (safety systems) should be implemented to prevent and mitigate the radiological consequences of DBAs through the prevention of significant fuel damage and damage to the containment boundary in order to limit the radiological consequences to the public and the environment to the extent that no special measures are required for the protection of the public.	To include the full SSR-2/1 quotation and emphasise prevention as well as mitigation.	X	3.13 Paragraph 5.25 of SSR-2/1 (Rev. 1) [1] states: "The design shall be such that for design basis accident conditions, key plant parameters do not exceed the specified design limits. A primary objective shall be to manage all design basis accidents so that they have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site protective actions." Consequently, specific design provisions (i.e. safety systems) should be implemented to prevent and mitigate the radiological consequences of design basis accidents by preventing significant fuel damage and maintaining the integrity of the containment (i.e. by preserving the structural integrity of the containment and maintaining its associated systems). The objective of the safety systems is to limit the radiological consequences for the public and the environment to the extent that no additional safety features or off-site protective		
111.	Canada	16	3.10	2nd para	*Consequently, specific design provisions (safety systems) should be implemented to mitigate the*	To improve the clarity.				

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					radiological consequences of DBAs through the prevention of significant fuel damage and ~~damage to~~ maintain the containment ~~boundary~~ functional integrity in order to …			actions are necessary for the protection of the public.		
112.	Germany	19	3.10	Last sentence	… Consequently, specific design provisions (safety systems) should be implemented to mitigate the radiological consequences of DBAs through the prevention of significant fuel damage, isolation of the containment atmosphere and avoidance of a pressure and temperate increase in the long term which may damage ~~to~~ the containment boundary in order to limit the radiological consequences to the public and the environment to the extent that no special measures are required for the protection of the public.	Usually, on level 3 of defence in depth the objective is to isolate the containment atmosphere from the environment to ensure confinement of radioactive material released into the containment. For DBA, the main risk challenging the containment integrity will be a temperature / pressure built up in the long term. This would also ensure consistency with para 3.12.		Text added as follow and footnote with reference to SSG-53 was added. 3.13 …Consequently, specific design provisions (i.e. safety systems) should be implemented to prevent and mitigate the radiological consequences of design basis accidents by preventing significant fuel damage and maintaining the integrity of the containment (i.e. by preserving the structural integrity of the containment and maintaining its associated systems). The objective of the safety systems is to limit the radiological consequences for the public and the environment to the extent that no additional safety features or off-site protective actions are necessary for the protection of the public.		
113.	ENISS	19	3.11		The *operation* of safety systems designed to control DBAs with a quick development should rely on automatic actuation and should not involve short term human intervention for a sufficiently long period of time….	Clarification of the acceptability of human actuation of safety systems. Automatic actions are adequate for fast developing events in the reactor but may be seen as highly demanding for spent fuel pool events or even for some shutdown states.		3.14 …The set of postulated initiating events considered for design basis accidents should cover all challenges to the safety functions and barriers with which the safety systems are designed to cope. Safety systems designed to control design basis accidents should rely on automatic actuation and should avoid the need for short term operator actions.		There is no need to specify the kinetics of the DBA.
114.	ENISS	20	3.11		Design basis accidents are postulated events that are not expected to occur during the lifetime of the plant. The most frequent events categorized as DBAs should have an expected frequency typically below 10-2 per reactor-year. DBAs should include both hypothetic and rare single initiating events and also frequent single initiating events that failed to be controlled in the second level of defence in depth.	Clarification. In this description of DID implementation through the provision of equipment/provision, the text should be about reinforcing barriers not on postulating	X	3.14 … Safety systems should be designed and constructed as well as maintained to ensure sufficient reliability. Safety design concepts, such as conservative safety margins and redundancy, are required to be applied in their design and construction, and the environmental conditions considered in their qualification programme should correspond to the loads and adverse environmental conditions induced by design basis accidents, postulated internal and external hazards. Text deleted		

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					The operation of safety systems designed to control DBAs with a quick development should rely on automatic actuation and should not involve short term human intervention for a sufficiently long period of time and ~~their reliability should be very high~~ they should perform their safety functions with an adequate reliability level. Safety systems should be designed to ensure their ~~reliable operation~~ required safety functions under postulated external hazards and prevailing environmental conditions. The reliability level of safety systems should be such that the collective contribution to the core damage frequency of the failure probabilities to control DBAs does not exceed the quantitative safety goals of the plant (for new nuclear power plants typically below 10-5 per reactor-year). If this is not the case, dedicated design features should be considered with a reliability level appropriate to achieve such goals. These additional lines of protection should be assessed as DEC.	faults. See suggestion to clarify the text in that perspective.				
115.	WNA	1	3.11		DBAs should include both rare single initiating events and also frequent single initiating events that failed to be controlled in the second level of defence in depth. The extent of those PIE should be such that the safety systems are designed to cope with any kind of elementary challenge to a safety function, up to serious ones. The operation of safety systems designed…	As DEC are made of "unlikely yet credible single or multiple failures with the potential for exceeding the capabilities of safety systems designed for the mitigation of DBAs", the border between DBA and DEC depends on DBA definition. It should be ambitious enough so that safety systems are able to cope with any elementary challenge to a safety function	X	3.14 Design basis accidents are originated by postulated initiating events that are not expected to occur during the lifetime of the plant. The most frequent accidents categorized as design basis accidents should have an expected frequency typically below 10-2 per reactor-year. Design basis accidents should include both rare single initiating events and frequent single initiating events that failed to be controlled at the second level of defence in depth. The set of postulated initiating events considered for design basis accidents should cover all challenges to the safety functions and barriers with which the safety systems are designed to cope.		
116.	France	27	3.11		Design basis accidents are postulated events that are not expected to occur during the lifetime of the plant. The most frequent events categorized as DBAs should have an expected frequency below 10-2 per reactor-year. The operation of safety systems designed to control DBAs should rely on automatic actuation and should not involve human intervention for a sufficiently long period of time ~~and their reliability should be very high~~. Safety systems should be designed to ensure their reliable	This does not provide guidance: very high reliability is expected for many SSCs important to safety		3.14 …Safety systems designed to control design basis accidents should rely on automatic actuation and should avoid the need for short term operator actions. Safety systems should be designed and constructed as well as maintained to ensure sufficient reliability. Safety design concepts, such as conservative safety margins and redundancy, are required to be applied in their design and construction, and the environmental conditions considered in their qualification programme should correspond to the loads and adverse environmental		

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					operation under postulated external hazards and prevailing environmental conditions. The reliability of safety systems and DEC A provisions should notably be such that the ~~collective contribution to the~~ estimated core damage frequency ~~of failing to control DBAs~~ does not exceed the safety goals of the plant (for new nuclear power plants typically below 10-5 per reactor-year). ~~If this is not the case, DEC without significant fuel degradation should be postulated for specific low frequency sequences as appropriate to achieve such goals.~~	This is not understandable: DBA provisions should be sufficiently reliable and yet DEC A provisions should be implemented anyway. Moreover reliability of systems is not only based on probabilistic calculation. Reliability is not only a quantitative probabilistic concept. Consideration of SFC enhances reliability of a system without consideration of probabilistic analysis.		conditions induced by design basis accidents, postulated internal and external hazards. Further recommendations on the design of specific safety systems for nuclear power plants are provided in the corresponding Safety Guides [5-8].		
117.	Canada	17	3.11	7th line	*Safety systems should be designed to ensure their reliable operation under ~~postulated external hazards and prevailing environmental conditions~~ the loads and adverse environmental conditions induced by design basis accidents and/or postulated internal or external hazards (e.g., seismic event, fire, internal flooding, etc.) through qualification and/or protection.*	Internal hazards should also be included. Other changes to improve clarity.				
118.	Germany	20	3.11		Safety systems should be designed to ensure their reliable operation under postulated internal and external hazards and prevailing environmental conditions.	Safety systems have to fulfill their function also in case of internal hazards. These has to be ensured by an appropriate design in such a way, that an internal hazard will only affect one redundant train of the safety system.				
119.	WNA	2	3.11		The reliability of safety systems should be such that the collective contribution to the core damage frequency of failing to control DBAs does not exceed the safety goals of the plant (for new nuclear power plants typically below 10-5 per reactor-year). If this is not the case, DEC without significant fuel degradation should be postulated for specific low frequency sequences as appropriate to achieve such goals. For this purpose, the overall CDF target should be refined into a core melt prevention frequency target for each single PIE.	A balanced design requires that core melt can be prevented with high confidence for any PIE			X	Original text was deleted. Proposed text is rejected since providing recommendations on probabilistic safety goals is out of the scope of this safety guide. Therefore, the proposed text is not necessary.
120.	Canada	18	3.11	9th line	*~~The reliability of safety systems should be such that the collective contribution to the core damage frequency of failing to control DBAs does not exceed the safety goals of the plant (for new nuclear power plants typically below 10-5 per reactor-year). If this is not the case, DEC without significant fuel degradation should be postulated for specific low frequency sequences as appropriate to achieve such goals.~~*	Safety systems are systems used in mitigating DBAs or level 3 DiD. As such, they should ensure that DBAs meet dose acceptance criteria for events with a frequency of more than 10^{-5}. The reliability of safety systems should be set high. Paragraph 3.46 provides the adequate requirement, no need for this one.				Original text was deleted.

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121.	UK	18	3.11		Delete text "~~The reliability of safety systems should be such that (to the extent possible) the collective contribution to the core damage frequency of failing to mitigate DBAs does not exceed the safety goals of the plant (for new nuclear power plants typically below 10- 5 per reactor-year). If this is not the case, DEC without significant fuel degradation could be postulated for specific low frequency sequences as appropriate to achieve such goals.~~"	A repeat of comment made by the UK at Step 7 prior to MS comment stage: In the UK, the consideration of DBAs is principally a deterministic matter. The second half of para 3.11 changes from deterministic expectations for DBAs to PSA expectations: *"The reliability of safety systems should be such that (to the extent possible) the collective contribution to the core damage frequency of failing to mitigate DBAs does not exceed the safety goals of the plant (for new nuclear power plants typically below 10- 5 per reactor-year). If this is not the case, DEC without significant fuel degradation could be postulated for specific low frequency sequences as appropriate to achieve such goals."* Safety systems should be very reliable, but this should be driven by deterministic rules (design codes, SSC classification etc) as well as PSA. In addition, the text above seems to suggest that if a design has very reliable safety systems, DEC without core damage may not need to be considered – DEC without core damage are only needed if a CDF target cannot be met without them. SSR2/1 (as quoted in para 3.13) states DECs should be identified on the basis of "engineering judgement, deterministic assessments and probabilistic assessments". PSA is just one aspect. It seems unlikely that for any current NPP technology, safety systems for DBAs could be so reliable that DEC without core damage never need to be considered. The conditions for DEC without core damage are set out in para 3.17, and para 3.23 talks about how DECs can reduce the frequency of severe accidents caused by failures of DBA measures. The statement at the end of para 3.24 makes a similar point but is more general ie "*The reliability of safety systems should be high enough for DEC without significant fuel degradation to only be postulated exceptionally and to occur with a frequency.*" **In summary - Propose deleting text from 3.11 as points are covered elsewhere in a more acceptable way.**			X	Original text was deleted.

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122.	Germany	21	3.11	Last sentence	… ~~If this is not the case, DEC without significant fuel degradation should be postulated for specific low frequency sequences as appropriate to achieve such goals.~~	It cannot be an acceptable approach for new NPPs to design for core damage frequencies higher than 10^{-5} per reactor-year. Hence, this Safety Guide should not deal with exceptions from the requirement in the sentence before.	X			
123.	France	28	3.12		If the design of the containment is such that in the case of the most limiting DBAs the intervention of cooling or pressure reduction systems (e.g. containment spray) is necessary to ensure the integrity of the containment boundary, such systems should be designed, constructed and maintained to ensure a ~~very high~~ reliability commensurate with the consideration that, ~~since~~ their failure would not only lead to a severe accident but also jeopardize the subsequent measures for its mitigation. . ~~For the same reason,~~ containment isolation provisions in case of DBAs should also be designed to have very high sufficient reliability for ensuring that acceptable limits for radiological consequences are not exceeded ~~and sufficient coolant inventory can be~~.	We might live with the first sentence but there is no really guidance High reliability: see above This provides no guidance, containment isolation is not designed regarding loss of coolant inventory				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.
124.	WNA	3	3.12		If the design of the containment is such that in the case of the most limiting DBAs the intervention of cooling or pressure reduction systems (e.g. containment spray) is necessary to ensure the integrity of the containment boundary, such systems should be designed, constructed and maintained to ensure a very high reliability, since if their failure would not only lead to a severe accident but also jeopardize the subsequent measures for its mitigation. Preferably, diverse means should be designed for this purpose in accordance with the independence requirements between levels of defense.	The IAEA should not encourage a design that does not fulfil the independence requirement between levels of DID				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.
125.	ENISS	21	3.12		If the design of the containment is such that in the case of the most limiting DBAs the intervention of cooling or pressure reduction systems (e.g. containment spray) is necessary to ensure the integrity of the containment boundary, such systems should be designed, constructed and maintained to ensure a sufficient ~~very high~~ reliability and redundancy, since their failure would not only lead to a severe accident but also jeopardize the subsequent measures for its mitigation. For the	The point here is to say that the containment is used for DBA and DEC, and therefore should not be lost in DBA to ensure the confinement safety function in DEC. A "very high reliability" is required, where SSR-2/1 is requiring "sufficient reliability". Consider alignment with requirement 6.28 of SSR-2/1 along with 6.28A as a complement:				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.

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					same reason, containment isolation provisions in case of DBAs should also be designed to have sufficient ~~very high~~ reliability for ensuring that acceptable limits for radiological consequences are not exceeded and sufficient coolant inventory can be maintained	"*6.28. The capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain them at acceptably low levels after any accidental release of high energy fluids.* *The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled.*" "*6.28A. Design provision shall be made to prevent the loss of the structural integrity of the containment in all plant states. The use of this provision shall not lead to an early radioactive release or a large radioactive release.*"				
126.	Japan	2	3.12		If the design of the containment is such that in the case of the most limiting DBAs the intervention of cooling or pressure reduction systems (e.g. containment spray) is necessary to ensure the integrity of the containment boundary, such systems should be designed, constructed and maintained to ensure a very high reliability, since their failure would not only lead to radioactive releases ~~a severe accident~~ but also jeopardize the subsequent measures for its mitigation. For the same reason, containment isolation provisions in case of DBAs should also be designed to have very high reliability for ensuring that acceptable limits for radiological consequences are not exceeded and sufficient coolant inventory can be maintained.	Failure of cooling or pressure reduction systems for containment does not always lead to a severe accident. Those failures result in loss of function to confine radioactive materials.				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.
127.	Canada	19	3.12	4th line	*If the design of the containment is such that in the case of the most limiting DBAs the intervention of cooling or pressure reduction systems (e.g. containment spray) is necessary to ensure the integrity of the containment boundary, such systems should be designed, constructed with a significant conservative margin and maintained to ensure a very high reliability, ~~since their failure would not only lead to a severe accident but also jeopardize the subsequent measures for its mitigation,~~ considering they are required to operate under severe accident conditions to protect the containment integrity from the challenges imposed by severe core damage phenomena.*	To improve the accuracy and clarity.				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.

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128.	India	6	3.12	Line 5	If the design of the containment is such that in the case of the most limiting DBAs the intervention of cooling or pressure reduction systems (e.g. containment spray) is necessary to ensure the integrity of the containment boundary, such systems should be designed, constructed and maintained to ensure a very high reliability, since their failure ~~would~~ may lead to a severe accident but also jeopardize the subsequent measures for its mitigation.	Containment pressure reduction system failure will not lead to Severe accident but it may lead to an early radioactive release or a large radioactive release. In certain cases, it may lead to severe accident				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.
129.	Germany	22	3.12	2nd sentence	… For the same reason, containment isolation provisions in case of DBAs should also be designed and maintained to have very high reliability for ensuring that acceptable limits for radiological consequences are not exceeded and sufficient coolant inventory can be maintained.	Containment isolation provisions such as valves need maintaining, which should be included.				Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.
130.	UK	19	3.12		Suggest this paragraph is deleted	Having introduced generic expectations for DBA safety systems in para 3.11, this paragraph talks about specifics of containment systems, essentially confirming what has already been said, i.e. that measures for DBAs should be 'highly reliable'. Given that this guide is not focussing on DBAs, this detail seems unnecessary.	X			Original text deleted since generic expectations for safety systems required in DBA are introduced in para 3.14.
131.	France	29	3.16 and 3.17		Consider replacement of this articles by : "SSG-2 articles 3.39 and 3.40 provide guidance regarding development of "deterministically derived list of design extension conditions without significant fuel degradation … " + exact quotation of these articles	Consistency with SSG-2 shall be ensured and 3.16/17 deal with exactly the same topic as these articles of SSG-2.	X	3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
132.	Canada	20	3.16		Delete this paragraph as the topic is already covered SSG-2. If it is retained, then the text should be corrected.	SSG-2 already provides guidance for identification, grouping, classification and analysis of event sequences. The implication that events for a particular plant state are identified independently of events in other plant states is incorrect. Identification comes before classification. Furthermore, is not clear from the wording of SSR-2/1 Requirement 20 that the identification of DEC scenarios for study is intended to be "comprehensive". It can equally be read as requiring the identification of many scenarios, but study (analysis) of a limited set of bounding scenarios. This is the interpretation used in SSG-2.		3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
133.	India	7	3.16	Line 3	AOOs and ~~the most frequent~~ DBAs which have higher frequency of occurrence combined with a common cause failure on redundant	For clarity and consistency with clause 3.17 b		Original text deleted and references to paras. From SSG-2 (Rev.1) were added		

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					equipment from a safety system are expected to provide most of such credible conditions.			3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
134.	UK	20	3.16		Missing full-stop at end of paragraph.	Minor typographical	X			
135.	Canada	69	3.17 **Major comment**		See general comment above the table	For classification of plant states, some member states (e.g. Canada) do not classify DEC-A and DEC-B separately – they are all DEC. At the time of classification, the results of the analysis are not known, therefore DEC-A and DEC-B cannot be distinguished. It is recognised that iteration between design and analysis will resolve much of this problem. But consider: for States using DiD approach 1: 1. A scenario allocated to DEC-A will be analysed using conservative DBA methods and may fail to prevent core melt, so it must be DEC-B. 2. For the next iteration it will be analysed with best-estimate methods and may show no core damage. So it must be DEC-A 3. Go back to 1 There are problems with using the output from safety analysis in the classification of scenarios, which is an input to the safety analysis.		Original paragraph deleted. 3.17 To meet the requirements presented in paras 3.15 and 3.16, two separate categories of design extension conditions should be identified: design extension conditions without significant fuel degradation and design extension conditions with core melting. 3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
136.	WNA	4	3.17		An initiating event less frequent than those considered for DBAs, with a frequency in the same order of magnitude as the core melt prevention target and that exceeds the capabilities of safety systems for mitigation of DBAs;	DBA should remain the preferable plant state to deal with single initiating events, as it is the basis for the design of the safety systems. Otherwise safety systems are only designed to mitigate frequent initiating events. DEC rules should only be acceptable for single PIE that are so rare that you may even wonder if it is worth analysing them as they are close to the residual risk.	X			
137.	Canada	21	3.17		*b) An anticipated operational occurrence or frequent design basis accident combined with the failure of a safety system or safety systems designed for its mitigation~~, typically~~ due to a common cause failure or other failure;*	It is important to note that more than one failure can occur in safety systems and that they are not only caused by common cause failures				

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					c) A postulated initiating event associated with the failure of a safety system used for normal operation (eg. A support system that is required for the control of the initiating event) or required for the mitigation of AOO and/or DBA.					
138.	Japan	3	3.17		Design extension conditions without significant fuel degradation are to a large extent technology and design dependent, but they can be classified in three types [8], as follows: (a) An initiating event less frequent than those considered for DBAs and that exceeds the capabilities of safety systems for mitigation of DBAs; (b) An anticipated operational occurrence or frequent design basis accident combined with the failure of a safety system designed for its mitigation, typically due to a common cause failure; (c) A postulated initiating event associated with the failure of a safety system used for normal operation, e.g. a ~~support system~~ safety system support feature, that is required for the control of the initiating event.	Support system in safety system is defined as "safety system support feature" in IAEA Safety Glossary 2018.		Original paragraph deleted. 3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
139.	Canada	23	3.17 (a) to (c)		Provide design-neutral examples where possible.	Examples would improve understanding.		Original paragraph deleted. 3.17 To meet the requirements presented in paras 3.15 and 3.16, two separate categories of design extension conditions should be identified: design extension conditions without significant fuel degradation and design extension conditions with core melting. 3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
140.	India	8	3.17	(b)	'Anticipated operational occurrences ~~or~~ *and* frequent design basis accidents combined...'	To be consistent with clause 3.16.				
141.	Canada	22	3.17		Modify text to include the possibility of using the full design capability of the plant. Also reference the possibility of use of non-permanent equipment in the long-term management of an accident.	Clearly the emphasis is on dedicated safety features for DEC or extension of safety systems. But SSR-2/1 also allows crediting the use of non-safety or temporary equipment. See footnote 15 to SSR-2/1 para 5.29. Text should be revised to reflect this. "*15 For returning the plant to a safe state or for mitigating the consequences of an accident, consideration could be given to the full design capabilities of the plant and to the temporary use of additional systems.*" See also SSG-2 paragraphs 7.51 and 7.64.				

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142.	India	9	3.17	(c)	*Suggestion* The example at the end of the clause does not adequately reflect the intent of the clause. The example may be re-considered for clearly bringing out the intent.	There is a contradiction within this Clause, it starts with postulation of failure of a "safety system", but ends by giving example of a support system. Change is suggested to bring in the exact intended meaning of the applicable Clause 3.40 (c) of SSG-2. Therein, SSG-2 gives example of a heat removal system which qualifies more as a safety system rather than just a support system.		Original paragraph deleted. 3.17 To meet the requirements presented in paras 3.15 and 3.16, two separate categories of design extension conditions should be identified: design extension conditions without significant fuel degradation and design extension conditions with core melting. 3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations for the identification of design extension conditions without significant fuel degradation.		
143.	India	10			*Page 11 Suggestion* Similar para as 3.27 may be included in the section on 'DEC without significant fuel degradation'	For clarity on the requirements and ensuring availability and survivability of the additional safety systems to cover DEC with confidence.	X			
144.	Canada	70	3.18 **Major comment**		Change the beginning of the paragraph as follows. "*The objective for DBA in SSR-2/1 paragraph 5.25 is to have no or only minor radiological consequences such that off-site protective actions are not needed.* *The objective for DEC in SSR-2/1 §5.31A is to limit radiological consequences such that necessary off-site protective actions are limited in time and area.* *Some Member States may choose to apply the DBA requirement to DEC-A. For those member states, the following guidance may be applied:* ~~*The objective in DBA and in DEC without significant fuel degradation is the same*~~*, namely to prevent core damage …*"	This is incorrect. The objectives are not the same for DBA and DEC. For DBA, see SSR-2/1 § 5.25. For DEC (both DEC-A and DEC-B) see SSR-2/1 §5.31A. See general Comment #1 above for more detail. Fuel degradation is not precluded in either plant state. Objectives for <u>all of DEC</u> are to: • prevent significant fuel damage to the extent practicable • when significant fuel damage occurs, mitigate the release Beyond the first part of the first sentence, we could not understand what the paragraph was trying to say. The remaining text should be edited for clarity.		3.18 Text modified as: 3.19 In general, the mitigation of design extension conditions without significant fuel degradation should be accomplished by safety features specifically designed and qualified for such conditions. Alternatively, design extension conditions without significant fuel degradation can be mitigated by available safety systems that have not been affected by the events that led to the design extension conditions under consideration and that are capable and qualified to operate under the associated environmental conditions. A difference between design basis accidents and design extension conditions without significant fuel degradation is established in some States in terms of their frequencies of occurrence. Very low frequency initiating events are treated as design extension conditions without significant fuel degradation. In other States, design extension conditions without significant fuel degradation are postulated for complex sequences involving multiple failures, whereas very low frequency postulated initiating events are treated as design basis accidents.		
145.	France	30	3.18 and 3.19		These articles should be improved to provide guidance and added value regarding SSG-2. If not, straightforward quotation of SSG 2 is sufficient. by quotation of SSG-2 articles 7.45 – 7.55	SSG-2 articles provide more complete guidance and deal with exactly the same topic. Replacement will ensure consistency and would provide guidance		3.19 text modified as: 3.20 The safety analyses of design basis accidents and design extension conditions without significant fuel degradation share similar safety objectives, namely, to maintain the integrity of		

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								barriers and to prevent core damage or damage to the fuel in the spent fuel pool (see paras 7.28 and 7.45 of SSG 2 (Rev. 1) [9]).		
146.	Canada	71	3.19	Last sentence	Change “high” to “adequate”. “3.19 … *There should still be ~~high~~ adequate confidence in the results and the margins to avoid cliff-edge effects should be demonstrated to be adequate*.”	I can find no basis for this statement in SSR-2/1 or SSG-2. SSG-2 §7.46 specifically has an objective of an “adequate level of confidence” not “high confidence”. Later paragraphs of SSG-2 do not require application of SFC and allows best-estimate in the analysis and best-estimate operator action assumptions. This is consistent with a lower level of confidence than the “high level” required for DBA.		3.22 …Nevertheless, there should still be adequate confidence in the results of the safety analysis and the safety margins to avoid cliff edge effects should be demonstrated to be adequate (see paras 7.54 to 7.55 of SSG 2 (Rev. 1) [9]).		
147.	Indonesia	3	3.19	7-8	Since the objective in DBA and in DEC without significant fuel degradation is the same, namely to prevent core damage or damage to the fuel in the irradiated fuel storage, the primary difference between these two accidental conditions is the use of different acceptance criteria, different design requirements or different approaches for performing safety analysis for this objective. Thus, in design extension conditions the following apply: Less stringent design requirements than for DBA can be applied, for example compliance with the single failure criterion is not required, equipment ~~can have~~ having a lower safety class and less rigorous reliability measures are allowed	Replace can have with having to make Para 3.19 grammatically correct.		3.21 …(a) Less stringent design requirements than for design basis accidents might be applied: for example, safety features for design extension conditions without significant fuel degradation may be assigned to a lower safety class than safety systems; the single failure criterion is applied at the function level (i.e. functional redundancy) but is not applied at the system level (i.e. no redundancy among systems is applied); and supporting systems (e.g. cooling system) and I&C systems (e.g. the signal for anticipated transients without scram) may be more diversified than supporting systems and I&C systems used for design basis accidents;		
148.	Canada	24	3.19		*~~Since the~~ The objective in DBA and in DEC without significant fuel degradation is the same, namely to prevent core damage or damage to the fuel in the irradiated fuel storage~~,~~. ~~the~~ The primary difference between these two accidental conditions is the use of different levels of conservative in acceptance criteria, ~~different~~ design requirements or performing safety analysis for this objective, because of significant differences in the likelihood of their occurrence*.	To improve the clarity	X	3.19 text modified as: 3.20 The safety analyses of design basis accidents and design extension conditions without significant fuel degradation share similar safety objectives, namely, to maintain the integrity of barriers and to prevent core damage or damage to the fuel in the spent fuel pool (see paras 7.28 and 7.45 of SSG 2 (Rev. 1) [9]). And text: 3.21 Design basis accidents and design extension conditions without significant fuel degradation are also distinguished in terms of the application of different design requirements, and the use of different acceptance criteria or approaches for performing safety analysis. Thus,		
149.	Germany	23	3.19		Since the objective in DBA and in DEC without significant fuel degradation is the same, namely to prevent core damage or damage to the fuel in the irradiated fuel storage, the primary difference between these two accidental	It is not a matter of fact that there are differences. This is up to the national practice. In the following bullet the wording “can be applied” is not clear. Again this is up to the national practice				

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					conditions may be ~~is~~ the use of different acceptance criteria, different design requirements or different approaches for performing safety analysis for this objective. Thus, in design extension conditions the following apply: (a) Less stringent design requirements than for DBA ~~can~~ may be applied, for example compliance with the single failure criterion is not required, equipment can have a lower safety class (SSG 30, para 3.13) and less rigorous reliability measures are allowed; (b) Less conservative assumptions and criteria than for DBA, or best estimate methods, are acceptable for the safety analysis (SSG 2, (Rev. 1) para 7.54).	and therefore "may be applied" has to be used, see e.g. in para. 3.20. Please add references for clarification.		for design extension conditions without significant fuel degradation the following apply: (a) Less stringent design requirements than for design basis accidents might be applied: for example, safety features for design extension conditions without significant fuel degradation may be assigned to a lower safety class than safety systems; the single failure criterion is applied at the function level (i.e. functional redundancy) but is not applied at the system level (i.e. no redundancy among systems is applied); and supporting systems (e.g. cooling system) and I&C systems (e.g. the signal for anticipated transients without scram) may be more diversified than supporting systems and I&C systems used for design basis accidents; (b) Less conservative assumptions than for design basis accidents, or best estimate methods, are acceptable for the safety analysis (see paras 7.35 to 7.44 and 7.47 to 7.55 of SSG-2 (Rev. 1) [9]); (c) The acceptable criteria related to the radiological consequences for design extension conditions without significant fuel degradation may be identical or similar to those for design basis accidents (see paras 7.32 to 7.33 and 7.46 of SSG-2 (Rev. 1) [9]).		
150.	UK	21	3.19		Delete the first sentence. Second sentence to read: "For consideration of design extension conditions, the following apply:"	Is this first sentence really necessary, given that this is more succinctly summarised in (a) & (b) ?		3.21 Design basis accidents and design extension conditions without significant fuel degradation are also distinguished in terms of the application of different design requirements, and the use of different acceptance criteria or approaches for performing safety analysis. Thus, for design extension conditions without significant fuel degradation the following apply:…		
151.	Canada	72	3.20,	editorial .	Change 3.17 (a) to 3.18 (a).	Reference in 3rd sentence is incorrect	X	Paragraph numbers changed after edition		
152.	France	31	3.20		Consider deletion	This article explain that the use of safety systems is – by nature – a good practice for DEC A, simply because use of systems designed with stringent rules leads automatically to achieve criteria when rules are less stringent…		3.22 If it is possible to use available safety systems to respond to design extension conditions without significant fuel degradation, safety analysis is still required to demonstrate their effectiveness: see Requirement 42 of SSR-2/1 (Rev. 1) [1]. The safety analysis may use less conservative methods and assumptions than for design basis accidents (otherwise there would be no differentiation between design basis accidents and design extension conditions without significant fuel degradation). Nevertheless, there should still be adequate confidence in the results of the safety analysis and the safety margins to avoid cliff edge		Other MS propose a modification, which was accepted, and the text modified as presented.
153.	Canada	25	3.20		Remove paragraph.	This paragraph does not add anything to the document. If the emphasis was meant on ensuring sufficient margins to cliff edge, then it should be worded as such.				
154.	Japan	4	3.20		The use of available safety systems, when possible, in DEC without significant fuel degradation has the important advantage that safety systems are designed with very stringent	Clarification for plant states.				

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					reliability criteria. In such cases, the rules for safety analyses [8] use less conservative methods and assumptions but they should still ensure a high confidence in the results. Thus, when best estimate analysis is performed, the margins to avoid cliff edge effects should be demonstrated to be adequate. If the rules were the same, there would not be a need for differentiation between DBA and DEC without significant fuel degradation.			effects should be demonstrated to be adequate (see paras 7.54 to 7.55 of SSG 2 (Rev. 1) [9]).		
155.	UK	22	3.20		Change to: “Where it is possible to utilise available safety systems (provided primarily for DBAs) to respond to DEC without significant fuel degradation, safety analysis is still required to demonstrate their effectiveness. This analysis should use less conservative methods and assumptions than required for DBA (otherwise there would be no differentiation between DBA and DEC). There should still be high confidence in the results and the margins to avoid cliff-edge effects should be demonstrated to be adequate.”	Para 3.20 is currently not very clear. What is the ‘important advantage’ safety systems have? Is it trying to say that safety systems (for DBA) are usually designed and maintained to a higher standard than safety features dedicated to DECs? If so, say so. The next sentence starts with “In such cases”. Which cases? The previous sentence had been talking about safety systems, the second sentence is talking about analysis. Is the objective of the paragraph to say that DEC analysis demonstrating the effectiveness of DBA safety measures for DEC plant states still needs to be performed, but on a best estimate basis. This avoids the DBAs just being extended out to very low frequency events? If this is the case, the proposed text tries to give this clarity. Para 3.21& 3.24 talk about capability/qualification/reliability of DBA safety measures to work for DECs, therefore discussion of reliability is not proposed for 3.20.		3.22 If it is possible to use available safety systems to respond to design extension conditions without significant fuel degradation, safety analysis is still required to demonstrate their effectiveness: see Requirement 42 of SSR-2/1 (Rev. 1) [1]. The safety analysis may use less conservative methods and assumptions than for design basis accidents (otherwise there would be no differentiation between design basis accidents and design extension conditions without significant fuel degradation). Nevertheless, there should still be adequate confidence in the results of the safety analysis and the safety margins to avoid cliff edge effects should be demonstrated to be adequate (see paras 7.54 to 7.55 of SSG 2 (Rev. 1) [9]).		
156.	Canada	73	3.21		Quote the full text from SSG-34 para 5.8.	It would be better to quote the whole of SSG-34 §5.8. This would include the information that SBO does not include simultaneous failure of uninterruptible AC or DC power or diverse alternate AC sources that are not susceptible to the initiating event. SBO is often misinterpreted as loss of all electrical power or at least loss of all AC power. The suggested change would help prevent such a misinterpretation.	X	3.25 …In many plant designs, such conditions include anticipated transient without scram and station blackout, i.e. loss of the preferred power supply concurrent with a turbine trip and unavailability of all standby AC power supplies (see SSG-34 [7]).		
157.	Italy	13	3.21	1	[…] DECs without significant fuel degradation have the potential […]	Grammar (“have” is plural)	X			Considered during technical edition

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158.	Canada	26	3.21	last statement	*Therefore, for the conditions described in para. 3.172 (a) it may be possible to show that some safety systems with conservative margin embedded in their design would be capable of (and be qualified for) mitigating the event under consideration, based on best estimate analyses and less conservative assumptions.*	To improve the clarity		3.23 Therefore, for design extension conditions without significant fuel degradation it might be possible to show that some safety systems, with an extended capability embedded in their design, would be capable of, and be qualified for, mitigating the conditions under consideration, based on best estimate analyses and on less conservative assumptions than the assumptions used for design basis accidents.		
159.	Canada	27	3.22		*The plant management should have an understanding that ~~of~~ certain ~~the~~ security features of the nuclear power plant~~, as these that~~ might also be adversely affected by the impact of hazards or the necessary mitigation measures once activated, without protection or qualification.*	To improve the clarity.			X	This text doesn't belong to this guide, certainly not to 3.22. In addition, security considerations are out of the scope of this safety guide.
160.	Canada	28	3.22	last statement	*~~For the same reason, containment isolation provisions in case of DBAs should also be designed to have very high reliability for ensuring that acceptable limits for radiological consequences are not exceeded and sufficient coolant inventory can be maintained~~.*	The logic doesn't flow.	X	Original text is from paragraph 3.13 (not from 3.22). Original text was deleted.		
161.	Germany	24	3.22	Last sentence New footnote	… These include in many designs the anticipated transients without scram and station blackout[FN]. [FN] Understanding of the term station blackout is provided in para 5.8 of SSG-34.	Here, station blackout means unavailability of – off-site power – houseload operation – standby AC power sources This is clearly specified in para. 5.8 of SSG-34. It is proposed to add a reference to the para mentioned above, e.g. by a footnote.		3.25 …In many plant designs, such conditions include anticipated transient without scram and station blackout, i.e. loss of the preferred power supply concurrent with a turbine trip and unavailability of all standby AC power supplies (see SSG-34 [7]).		
162.	UK	23	3.22		Replace "station blackout" with "station blackout (defined in SSG-34 as loss of the preferred power supply concurrent with a turbine trip and unavailability of all standby AC power supplies)".	For accuracy/completeness				
163.	France	32	3.22		~~Design extension conditions should be considered for failures of safety systems designed both to cope with anticipated operational occurrences and DBAs. These~~ According to SSG-2 article 41, the list of DEC without significant fuel degradation includes in many designs the anticipated transients without scram and station blackout.	To ensure consistency with SSG-2, it is better to mention it when dealing with the same topic		3.25 Anticipated operational occurrences and design basis accidents combined with failures in safety systems should be considered as part of the list of design extension conditions without significant fuel degradation; see para. 3.40 of SSG-2 (Rev. 1) [9]. In many plant designs, such conditions include anticipated transient without scram and station blackout, i.e. loss of the preferred power supply concurrent with a turbine trip and unavailability of all standby AC power supplies (see SSG-34 [7]).		
164.	Canada	29	3.24		*Design extension conditions without significant fuel degradation contribute to achieve the*	To improve the clarity	X	3.27 Consideration of design extension conditions without significant fuel degradation		

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					fundamental objective of the design for DEC by taking all reasonable steps to reduce the chances that a DEC involving substantial damage to the reactor will occur. ~~a reinforcement of the design for some complex and unlikely failure sequences. As some safety systems are designed to cope with various DBAs (e.g. the emergency core cooling systems are designed for several sizes and locations of loss of coolant accidents or main steam line breaks), safety features for DEC can help to reinforce the capability of the plant for specific sequences improving and balancing the risk profile: applying less stringent design or safety assessment criteria than for DBA conditions could help to identify reasonably practicable provisions to improve safety. The reliability of safety systems should be high enough for DEC without significant fuel degradation to only be postulated exceptionally and to occur with a very low frequency.~~			reinforces the robustness of the design to cope with some complex and unlikely failure sequences and balances the overall risk profile of the plant. Therefore, the reliability of safety systems and safety features for design extension conditions without significant fuel degradation should be sufficiently high that escalation to a severe accident is very unlikely to occur.		
165.	Indonesia	4	3.24	9-10	Design extension conditions without significant fuel degradation contribute to a reinforcement of the design for some complex and unlikely failure sequences. As some safety systems are designed to cope with various DBAs (e.g. the emergency core cooling systems are designed for several sizes and locations of loss of coolant accidents or main steam line breaks), safety features for DEC can help to reinforce the capability of the plant for specific sequences improving and balancing the risk profile: applying less stringent design or safety assessment criteria than for DBA conditions could help to identify reasonably practicable provisions to improve safety. The reliability of safety systems should be high enough for DEC without significant fuel degradation to only be postulated exceptionally and to occur with a ~~very low~~ frequency less than considered for DBA	Delete very low and addless than to further clarify Para 3.24 and be consistent with Para 3.17		3.27 … Therefore, the reliability of safety systems and safety features for design extension conditions without significant fuel degradation should be sufficiently high that escalation to a severe accident is very unlikely to occur.		
166.	UK	24	3.24		Suggest deleting the last sentence "~~The reliability of safety systems should be high enough for DEC without significant fuel degradation to only be postulated exceptionally and to occur with a very low frequency~~."	The last sentence is not clear "The reliability of safety systems should be high enough for DEC without significant fuel degradation to only be postulated exceptionally and to occur with a very low frequency."		3.27 … Therefore, the reliability of safety systems and safety features for design extension conditions without significant fuel degradation should be sufficiently high that escalation to a severe accident is very unlikely to occur.	X	It is important to emphasize the reliability of safety systems and of safety features for design extension conditions without significant

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						Is it just talking about safety systems (for DBA) or safety features (for DEC), or both? If it is just talking about the reliability of safety systems for DBAs, which means that DECs should only occur with a low frequency, that point has already been made eg (at the end of the DBA section), and if it hadn't, it seems odd to leave it until right at the end of this section which is on DEC without fuel degradation. If it is saying that the reliability of safety systems/features for DECs (DEC without fuel degradation) should be high enough, such that escalation to DEC with ore melting is very unlikely, that is not clear.				fuel degradation in the prevention of severe accidents.
167.	France	33	3..25		In accordance with para. 5.30 of SSR-2/1 (Rev. 1) [1], on the basis of the up-to-date state of the art and status of R&D results, a set of ~~representative~~ accidents with core melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality.	"representative" is not really adequate in this context and this end of the sentence is sufficient regarding recommendation aspect. Regarding guidance aspects, mentioning state of the art (thus R&D is important regarding severe accidents). Note : it is also possible to introduce R&D in 3.26		3.28 In accordance with para. 5.9 of SSR-2/1 (Rev. 1) [1], and with consideration of results from research and development, a set of representative accident conditions with core melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality.		
168.	Indonesia	5	3.25	Line 1	In accordance with para. ~~5.30 of SSR-2/1 (Rev. 1) [1~~] para. 3.14, a set of representative accidents with core melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality. This set of accidents should be considered in the design of the corresponding safety features for DEC and should be a set of bounding cases that envelop other severe accidents with more limited degradation of the core. Significant fuel degradation in the irradiated fuel storage is not required to be included as a design extension condition in SSR 2/1. It is a plant conditions that should be practically eliminated.	Replace ~~5.30 of SSR-2/1 (Rev. 1) [1~~] with Para 3.14, since Para 3.25mentioned paragraph has been quoted in para 3.14 of the draft document				
169.	UK	25	3.25		This refers to 5.30 of SSR 2/1 on defining a set of 'representative' accidents which are 'bounding'. Is this the correct reference ? Should the reference here be to 5.9 of SSR 2/1 or possibly to (for example) 3.43 of SSG-2 ?)	Some operators may use the terms 'representative' and 'bounding' in different ways, so would be best to refer to correct IAEA terms/usage.	X	Text updated accordingly as: 3.28 In accordance with para. 5.9 of SSR-2/1 (Rev. 1) [1], and with consideration of results from research and development, a set of representative accident conditions with core melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality.		
170.	Germany	25	3.25		In accordance with para. 5.30 of SSR-2/1 (Rev. 1) [1], a set of representative accidents with core	To emphasize that accidents with significant fuel degradation in spent fuel storage have to be	X	3.29 Paragraph 6.68 of SSR-2/1 (Rev. 1) [1] states [footnote omitted]:		

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					melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality. This set of accidents should be considered in the design of the corresponding safety features for DEC and should be a set of bounding cases that envelop other severe accidents with more limited degradation of the core. ~~Significant fuel degradation in the irradiated fuel storage is not required to be included as a design extension condition in SSR 2/1. It is a plant conditions that should be practically eliminated.~~	practically eliminated, it is proposed to have a distinguished paragraph on this topic.		"For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site." Hence, significant fuel degradation in the spent fuel pool should not be postulated as part of this set of design extension conditions; rather it is required to be considered among the conditions to be practically eliminated (see Section 5).		
171.	Japan	5	3.25		In accordance with para. 5.30 of SSR-2/1 (Rev. 1) [1], a set of representative accidents with core melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality. This set of accidents should be considered in the design of the corresponding safety features for DEC and should be a set of bounding cases that envelop other severe accidents with more limited degradation of the core. Significant fuel degradation in the irradiated fuel storage is not required to be included as a design extension condition in SSR~~-~~2/1 (Rev. 1). It is a plant ~~conditions~~ states that should be practically eliminated.	Editorials.	X			
172.	Canada	30	3.25		Use a direct quote of SSR-2/1 para 5.30. Optional guidance following the quote should be prefaced by "*This may be achieved by ...*"	This is not what SSR-2/1 para 5.30 actually says. It does not mention representative accidents with core melting or design inputs. It seems like a reasonable interpretation, but it is an interpretation and other interpretations may be possible.		3.28 In accordance with para. 5.9 of SSR-2/1 (Rev. 1) [1], and with consideration of results from research and development, a set of representative accident conditions with core melting should be postulated to provide inputs for the design of the containment and of the safety features ensuring its functionality.		No need to quote but to make reference to correct para.
173.	WNA	5	3.25		Significant fuel degradation in the irradiated fuel storage is not required to be included as a design extension condition in SSR 2/1. ~~It is a plant conditions that should~~ if it is demonstrated to be practically eliminated	The design would be safer if fuel melt could be postulated in fuel storage pool. Practical elimination is never the preferred option.		3.29 Paragraph 6.68 of SSR-2/1 (Rev. 1) [1] states [footnote omitted]: "For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site." Hence, significant fuel degradation in the spent fuel pool should not be postulated as part of this set of design extension conditions; rather it is required to		

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								be considered among the conditions to be practically eliminated (see Section 5).		
174.	UK	26	3.25		Propose putting the last two sentences into their own new paragraph and expanding the text to include: "Severe accidents resulting in significant degradation of irradiated fuel in the spent fuel pool and other locations should be considered in the design of the facility and supporting safety analysis. However, given the focus of design extension condition analysis on maintaining the final containment barrier, on many designs it may not be the appropriate approach for demonstrating adequacy. Instead, plant conditions associated with significant degradation of irradiated fuel outside of main containment building (or where the containment is open/bypassed and cannot be closed in sufficient time) should be practically eliminated."	The last two sentences of the paragraph state effectively state that spent fuel pools do no need to be considered by analysis equivalent to DEC with core melting. This is a significant implication, that perhaps runs contrary to the learning from Fukushima that severe accidents can occur in spent fuel pools and they need to be analysed/protected against. As a result, it is recommended that these two sentences are expanded to give a longer explanation of what is expected and why.		3.29 Paragraph 6.68 of SSR-2/1 (Rev. 1) [1] states [footnote omitted]: "For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site." Hence, significant fuel degradation in the spent fuel pool should not be postulated as part of this set of design extension conditions; rather it is required to be considered among the conditions to be practically eliminated (see Section 5).		
175.	Germany	26	New para 3.25A		<u>Significant fuel degradation in the irradiated fuel storage is not required to be included as a design extension condition in SSR 2/1. It is a plant conditions that should be practically eliminated.</u>	We suggest separating of a current issue in a new para	X	3.29 Paragraph 6.68 of SSR-2/1 (Rev. 1) [1] states [footnote omitted]: "For reactors using a water pool system for fuel storage, the design shall be such as to prevent the uncovering of fuel assemblies in all plant states that are of relevance for the spent fuel pool so that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is 'practically eliminated' and so as to avoid high radiation fields on the site." Hence, significant fuel degradation in the spent fuel pool should not be postulated as part of this set of design extension conditions; rather it is required to be considered among the conditions to be practically eliminated (see Section 5).		
176.	Canada	31	3.26	4th line	*For new nuclear power plants, accidents involving core melting should be postulated as DEC, irrespective of the fact that the design provisions taken to prevent such conditions make the probability of core damage very low* <u>*(see also Section 4 for practical elimination of event sequences leading to early or large radioactive releases)*</u>	Guidance should be given on what is "new" for a nuclear power plant. Does it include Olkiluoto 3 EPR (construction started 2005)? Will it include EPR's based on the same design to be constructed in the future? Also, to improve clarity, .add text indicated at the end.	X	Original text related to "new NPP" was deleted. 3.30 … All accident conditions that could lead to core damage should be postulated as design extension conditions, even though the design provisions taken in accordance with the requirements of SSR-2/1 (Rev. 1) [1] to prevent such accidents will make the probability of core damage very low.		Proposed text to be added into () was not considered since it is mentioned in section 1, para 1.12.
177.	Indonesia	6	3.26	1-2	The accident conditions chosen should be justified based on engineering judgement see SSG-53 [5] and SSG-2 (Rev. 1) [8]) and insights from the probabilistic safety analyses: see see SSG-3 [9]~~) SSG-53 [5] and SSG-2 [8]~~. A detailed analysis should be performed and	Correction on the references.	X			

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					documented to identify and characterize accidents that can lead to core damage. For new nuclear power plants, accidents involving core melting should be postulated as DEC, irrespective of the fact that the design provisions taken to prevent such conditions make the probability of core damage very low. Aspects that affect the accident progression and that influence the containment response and the source term should be taken into account in the design of the safety features, as indicated in SSG-53 [5].					
178.	Canada	74	3.26		No change needed at this time.	This position appears reasonable for conventional NPPs with conventional fuel. Advanced designs, such as SMRs (which are outside the scope of SSR-2/1 and hence DS508) are beginning to challenge this position. Advanced fuel designs are under development that will challenge this position even for water-cooled NPPs. Soon SSR-2/1 (and so DS508) will need to deal with this. But for now, this can remain.				
179.	UK	27	3.26		Suggested text for 2nd sentence: “A detailed analysis should be performed and documented to identify and characterize accidents that can lead to core damage and also challenge or bypass the containment.”	It is stated that “A detailed analysis should be performed and documented to identify and characterize accidents that can lead to core damage.” However, it has previously been stated that DEC with core melting is focused on those events which can challenge the containment/final confinement barrier, and open containment/SFP accidents should be excluded. Therefore, should there be a statement about containment?		3.30 … A detailed analysis should be performed and documented to identify and characterize accident conditions that could lead to core damage and also challenge or bypass the containment.		
180.	India	11	3.26		The accident conditions chosen should be justified based on engineering judgement, ~~and~~ insights from the deterministic safety assessment, probabilistic safety analyses see SSG-53 [5] and SSG-2 [8].	The DSA, PSA insights are important here as highlighted in requirement of SSR 2/1. Further, reference to SSG-2 on ‘Deterministic Safety Assessment’ is made here.		3.30 The accident conditions chosen as design extension conditions with core melting should be justified on the basis of engineering judgement and insights from probabilistic safety analyses: see SSG-53 [6] and SSG-2 (Rev. 1) [9]….	X	Reference to para 5.30 of SSR-2/1 (Rev. 1) related to the accident conditions here considered. There the deterministic safety analyses are not considered.
181.	Indonesia	7	3.26	2	The accident conditions chosen should be justified based on engineering judgement and insights from the probabilistic safety analyses: see ~~SSG-53 [5] and SSG-2 [8].~~ SSG-2 (Rev. 1) [8] A detailed analysis should be performed and documented to identify and characterize	Correction on the references	X			

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					accidents that can lead to core damage. For new nuclear power plants, accidents involving core melting should be postulated as DEC, irrespective of the fact that the design provisions taken to prevent such conditions make the probability of core damage very low. Aspects that affect the accident progression and that influence the containment response and the source term should be taken into account in the design of the safety features, as indicated in SSG-53 [5].					
182.	Japan	6	3.26		The accident conditions chosen should be justified based on engineering judgement and insights from the probabilistic safety analyses: see SSG-53 [5] and SSG-2 (Rev. 1)[8]. ……… Aspects that affect the accident progression and that influence the containment response and the source term should be taken into account in the design of the safety features, as indicated in SSG-53 [5].	It should be specified the para. number for the referred safety guides. SSG-2 is superseded by SSG-2 (Rev. 1).	X			Reference can be made to SSR 2/1 (Rev.1), 5.30, which is the requirement that has been later on considered in other SGs
183.	France	34	3.28		The challenges to plant safety presented by DEC with fuel ~~core~~ melting, and the extent to which the design may be reasonably	Change also here "core melting" rather to "fuel melting". See 3.4.				This section is related to design extension conditions with core melting, therefore the text remains as proposed, see para 3.32 in final version.
184.	Japan	7	3.28		Recommendations in this regard are provided in IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [14].	It should be specified the para. number for the referred safety guides.			X	The part of SSG-54 related to severe accidents is all about this. There would be a huge and unnecessary list of paragraphs to include
185.	Italy	14	3.28	5	Recommendations in this regard are provided in IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [14].	Full stop missing at the end of the sentence.	X			Considered during technical edition
186.	France	35	3.29		Radioactive releases ~~due to leakage from~~ escaping the containment in a severe accident should remain below the safety limit ~~leak rate for sufficient time~~ to allow sufficient time for implementation of off-site protective actions. Beyond this time, containment ~~leakages~~ releases could exceed this limit but still be well below ~~the criterion for~~ the acceptable limit with protective actions in place and be well below a	Radioactive releases are estimated in mSv and should not be compared to a leakage rate estimated in mSv/hour. The release limit requiring offsite protective measures is defined in mSv. There is no requirement to establish such a criterion and this recommendation is not consistent with SSR-2/1 or with all practices.		3.33 Radioactive releases from the containment in a severe accident should remain below the safety limit to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases might exceed the safety limit but should still be well below the acceptable limits for design extension conditions with off-site protective actions in place. Radioactive releases should also be well below what is considered a		

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					large radioactive release. Moreover, according to SSG-53, "at the design stage, a target leak rate should be set that is well below the safety limit leak rate (i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions)". … … If a containment venting system is included in the design, ~~it should not be designed as the principal means of removing energy from the containment in case of severe accidents~~, it "should be used only as a last resort" and "multiple means to control the pressure buildup in accident conditions inside the containment should be implemented" according to SSG-53	Leaktighness of containment is delt in SSG-53 4.98 to 4.103: art 3.29 seems to be a downgrading of these articles, notably 4.100 that requires At the design stage, a target leak rate should be set that is **well below the safety limit leak** rate (i.e. **well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions**). The concept of principal means provides no guidance and is not technically understandable.. This article may also be a non useful rewording of objectives mentioned in SSR-2/1		large radioactive release. Moreover, as stated in para 4.100 of SSG-53 [6]: "At the design stage, a target leak rate should be set that is well below the safety limit leak rate (i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions)". This may be achieved by provision of adequate filtered containment venting or other design features or alternative measures.		
187.	ENISS	22	3.29		Radioactive releases ~~due to leakage~~ from the containment in a severe accident should remain below the safety criteria for this condition ~~limit leak rate for sufficient time~~ to allow for sufficient time to implement~~ation of~~ off-site protective actions. Beyond this time, containment releases ~~leakages~~ could exceed this short-term limit but still be below the acceptable long-term limit with protective actions in place and be well below the criterion for a large radioactive release.	Why is the focus on radiological leakages and not on releases? The point about protecting people is about the effective dose, as defined in ICRP documents. The "leakage rate" is depending on parameters such as the pressure inside the reactor building and may vary alongside the accident development. Crossing the leak rate for 1h may be acceptable in terms of doses to the public, especially where the leaks are collected. If a filtered venting system is used to reduce the containment pressure, the radiological consequences can't only be considered in terms of leakage, but this is the addition of the leakage releases and the filtered releases that has to be considered for the protection of the public that will be submitted to both. Radioactive releases are estimated in mSv and should not be compared to a leakage rate estimated in mSv/hour. The release limit requiring offsite protective measures is also defined in mSv.	X	3.33 Radioactive releases from the containment in a severe accident should remain below the safety limit to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases might exceed the safety limit but should still be well below the acceptable limits for design extension conditions with off-site protective actions in place. Radioactive releases should also be well below what is considered a large radioactive release. Moreover, as stated in para 4.100 of SSG-53 [6]: "At the design stage, a target leak rate should be set that is well below the safety limit leak rate (i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions)". This may be achieved by provision of adequate filtered containment venting or other design features or alternative measures.		

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						The practical measurement of the containment leakage rate is not easily achieved and is part of periodic tests. This requires a stable plant situation and no variations of pressure/temperature during the test, given the measurements uncertainties. It would not be practicable to measure a leakage rate in a severe accident condition. This has to be kept in mind to define requirements on containment leakage rate. Depending on the protective actions (sheltering only for example) there may be a need to follow a new limit if the public is not fully evacuated in a remote location.				
188.	Germany	27	3.29		Radioactive releases due to leakage from the containment in a severe accident should remain below the safety limit leak rate for sufficient time to allow implementation of off-site protective actions. Beyond this time, containment leakages could exceed this limit but must not exceed ~~still be well below~~ the criterion for a large radioactive release.	Clarification		3.33 Radioactive releases from the containment in a severe accident should remain below the safety limit to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases might exceed the safety limit but should still be well below the acceptable limits for design extension conditions with off-site protective actions in place. Radioactive releases should also be well below what is considered a large radioactive release. Moreover, as stated in para 4.100 of SSG-53 [6]: "At the design stage, a target leak rate should be set that is well below the safety limit leak rate (i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions)". This may be achieved by provision of adequate filtered containment venting or other design features or alternative measures.		
189.	UK	28	3.29		Proposed text for 1st & 2nd sentences: "Early radioactive releases should be limited to allow sufficient time for the implementation of off-site protective actions by ensuring the leakage from the containment in a severe accident is below an appropriate safety limit leak rate. Beyond this time, containment leakages could exceed this limit but they should still be well below the criterion for a large radioactive release".	With regards to the following text: "Radioactive releases due to leakage from the containment in a severe accident should remain below the safety limit leak rate for sufficient time to allow implementation of off-site protective actions. Beyond this time, containment leakages could exceed this limit but still be well below the criterion for a large radioactive release." Radioactive releases, leak rates and containment leakages are related but are not the same, and therefore one cannot be a limit or criterion for the other.				
190.	Canada	75	3.29		Suggest adding "*should*" for consistency with first sentence. Change 2 instances of "~~*well*~~ *below*" to *"below"*. "*3.29 Radioactive releases from the containment in a severe accident should remain below the safety limit to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases could exceed this limit but should still be ~~well~~ below the acceptable limit with protective actions in place and be ~~well~~ below a large radioactive release*."	There is no justification for requiring releases to be "well below" the safety limit. All that is required is to below the limit. That is what a limit means.		Text added and quotation of para in SSG-53 added. 3.33 Radioactive releases from the containment in a severe accident should remain below the safety limit to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases might exceed the safety limit but should still be well below the acceptable limits for design extension conditions with off-site protective actions in place. Radioactive releases should also be well below what is considered a large radioactive release. Moreover, as stated in para 4.100 of SSG-53 [6]:		

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								"At the design stage, a target leak rate should be set that is well below the safety limit leak rate (i.e. well below the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions)". This may be achieved by provision of adequate filtered containment venting or other design features or alternative measures.		
191.	ENISS	23	3.29		confinement ~~containment~~ function	Cf. IAEA 2018 Glossary	X			
192.	UK	29	3.29		Add to end of last sentence: "…early phases of the severe accident before off-site protective actions could be implemented."	To provide clarity on what is meant by "early phases"		3.33 Radioactive releases from the containment in a severe accident should remain below the safety limit to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases might exceed the safety limit but should still be well below the acceptable limits for design extension conditions with off-site protective actions in place. Radioactive releases should also be well below what is considered a large radioactive release. Moreover, as stated in para 4.100 of SSG-53 [6]:		Text proposed using "early phases of the severe accident" was deleted based on comments from emergency preparedness and response technical officers. New text is proposed instead.
193.	France	36	3.30		A safety assessment … More detailed information is provided in SSG-2 (Rev. 1) [8], notably regarding the examples of potential phenomena for LWR and influence of severe accident management strategy	The added value of this article regarding SSG-2 is not clear. At a minimum it is of high importance to highlight that the list is for LWR and is not always applicable, depending on the strategy		3.35 A safety assessment of the design should be performed with consideration of the progression of severe accident phenomena and their consequences, and the achievement of acceptable end state conditions and should take into account applicable topical issues. More detailed information on the range of physical processes that could occur following core damage is provided in para. 7.66 of SSG-2 (Rev. 1) [9].		Reference to para in SSG-2 (Rev. 1) is considered enough.
194.	Japan	8	3.30		More detailed information is provided in SSG-2 (Rev. 1) [8].	It should be specified the para. number for the referred safety guides.				
195.	India	12	3.30		Molten core re-location/ core-concrete interaction;	Core-relocation is an important aspect to be covered as part of the assessments				Text presented in 4.13, but the modification was not considered after technical edition
196.	UK	30	3.30		Consider replacing 'molten core' with 'molten corium' (e.g. as used in SSG-2) Move "Molten core stratification" down one place and change "criticality" to "re-criticality" (probably OK to keep first in list, or could also go lower down).	Use correct terminology Change list to order chronology and delineate different effects.	X			To be implemented where necessary
197.	Canada	32	3.30		Suggest adding the following additional items • *Rerelease and transport of fission products* • *Distribution of heat inside the reactor coolant circuit* *Elevated gas temperatures*	For the completeness of discussion			Y	Reference to para. 7.66 of SSG-2 (Rev.1) is provided instead in para 3.35. There is no need to repeat the text.

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198.	Egypt	2	3.30		Containment over temperature	Editorial issue				Not clear, but text revised.
199.	Italy	15	3.30	10	- Containment overpressurization	Typo (overpressurization has no space, optionally a hyphen)	X			Considered during technical edition
200.	India	13	3.30		New bullet may be included at the end of the bullets: Transport of radionuclides and aerosols in the containment	Transport of radionuclides in the containment is also an important theme from radiological impact/ consequence assessment point of view. Added bullet is consistent with SSG-2 (clause 7.66)				Text presented in 4.13, but the suggestion was not considered since the intention is to provide examples of plant event sequences that need to be considered for the implementation of the practical elimination concept.
201.	Canada	33	Section on Assessment of Implementation of the Defence in Depth Concept **Major comment**		Given the size and complexity of the document, Canada recommends that paragraphs 3.31 to 3.51 are deleted. Some material relevant to DEC and PE could be moved to other parts of the document if it is not there already.	This document has a main purpose of explaining DEC and PE, not providing a partial guide to DiD (it only addresses design aspects of DiD, does not cover Levels 1 and 5) and is limited to water-cooled NPPs. Defence in Depth is sufficiently important to need a full-scope Safety Guide of its own.				The consideration of DEC and PE in the frame of DiD was agreed and presented in the DPP. The WG of NUSSC has delimited the areas to be addresses in this safety guide. This comment contradicts these agreements. A new SG providing recommendations for the full implementation of DiD could be proposed to cover your expectations.
202.	France ENISS	37 24	3.31		The implementation of the concept of defence in depth~~, as implemented~~ in the design of a nuclear power plant, is required to be assessed to ensure that each level is adequately designed to meet its goals in terms of prevention, detection, limitation and mitigation	The original text may be understood as "an assessment of a concept". It may be better to clarify the intent.		3.36 The implementation of defence in depth in the design of a nuclear power plant is required to be assessed to ensure that the safety provisions for each level are adequately designed to meet the objectives of that level in terms of prevention, detection, limitation and mitigation.		
203.	UK	34	3.31-3.51		Suggest add sentence to start of 3.38: "The physical barriers included within a facility are an important consideration when assessing the adequacy of depth in depth implementation. For each identified source of radiation, the physical barriers (including the boundaries) should be identified and an evaluation of their robustness should be provided. The following	The section "ASSESSMENT OF THE IMPLEMENTATION OF THE DEFENCE IN DEPTH CONCEPT" is quite long and moves through a number of requirements without clear demarcation of a topic change. Para 3.34 focuses on "safety provisions for different plant states".	X	3.42 The physical barriers included in the design are an important consideration when assessing the adequacy of the implementation of defence in depth. For each identified source of radiation, the physical barriers (including the reactor coolant pressure boundary and the containment boundary) should be identified and their robustness should be evaluated in accordance		

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					aspects should be taken into account in the evaluation:"	Paras 3.35-3.37 consider levels of defence in depth. Para 3.38 is on physical barriers. Para 3.39 is on the performance of safety functions (in the context of safety provisions). Para 3.40 goes back to safety provisions. There is a danger as currently written that these terms could all be seen as equivalent. Physical barriers are clearly relevant to defence in depth, but even the 2005 Safety Report No.46 in its very first paragraph broadens out the concept from just barriers. Para 3.3(a) of DS508 identifies integrity of the barriers as a notable part of defence in depth but not the totality. This should be made clear around para 3.38.		with a graded approach. The following aspects should be assessed in the evaluation:		
204.	Japan	9	3.31 to 3.51		ASSESSMENT OF THE IMPLEMENTATION OF THE DEFENCE IN DEPTH CONCEPT *Insert subtitle such as,* *"Radioactive sources" for para 3.36 to 3.39* *"Deterministic safety analysis" for para 3.40 to 3.41* *"Probabilistic safety analysis" for para 3.42 to 3.-51*	User-friendliness.				Considered but not retained. Other subtitles were added instead.
205.	Italy	16	3.31	1	[…] nuclear power plant, has to be assessed to ensure that each level	Style (sentence is unnecessarily convoluted)	X			Considered during technical edition
206.	WNA	6	3.34		considering also all consequences of internal hazards and/or external hazards that could cause the event.	The specificity of hazards is that they can both cause a PIE and also cause additional failures that may affect the safety systems or safety features for DEC and thus disturb the safety demonstration.		3.38 …The assessment should demonstrate that, for each credible initiating event, the risk has been reduced to a level that is as low as reasonably achievable, considering also all consequences of internal hazards and external hazards that could cause the event.		
207.	Canada	76	3.34	2nd last sentence	"*It should demonstrate that, for each credible initiating event, the risk has been reduced as low as reasonably practicable, considering also ~~all~~ consequences of internal hazards and/or external hazards that could cause the event.*	The word "all" should be used very carefully. It allows no exceptions and is rarely achievable. Suggest deletion of "all".				
208.	UK	31	3.34		"The performance and reliability of safety provisions for different plant states should be assessed taking into consideration an applicable set of analysis rules, the level of risk and their safety significance."	Minor grammar comment	X			
209.	Egypt	3	3.35		SSCs of each level of defence are characterized by ~~a~~ reliability commensurate to their function and their significance.	Editorial issue.		3.39 … In a sound and balanced design, structures, systems and components at each level of defence are characterized by a reliability commensurate with their function and their safety		

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								significance, and reasonable safety margins are provided.		
210.	Canada	77	3.35		Suggest deletion of the paragraph or clarifying that independence is only required to the extent practicable.	Beware of over stating the need for complete protection at each level of DiD. Suggest deleting this paragraph or recognising that it is expected only "to the extent practicable". As acknowledged below in § 3.60, current NPP designs typically control a limited number of AOOs with safety systems (e.g. loss of flow accidents). Also note the following from SSR-2/1: • Requirement 7 only requires independence between levels "to the extent practicable" • § 4.11 (c) "failures and deviations from normal operation requiring actuation of safety systems are minimized or excluded by design, to the extent possible" § 4.13 "The design shall be such as to ensure, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the nuclear power plant."		Paragraphs 3.55 to 3.61 contain the explanation.		
211.	India	14	3.35		In a sound and balanced design, SSCs of each level of defence are characterized by a reliability commensurate to their function and their safety significance, providing reasonable safety margins.	The assessment of availability of adequate safety margins is an important aspect of sound design.	X	3.39 The multiplicity of the levels of defence is not a justification to weaken the effectiveness of some levels by relying on the effectiveness of other levels. In a sound and balanced design, structures, systems and components at each level of defence are characterized by a reliability commensurate with their function and their safety significance, and reasonable safety margins are provided.		
212.	Germany	28	3.36		The defence in depth strategy in the design of a nuclear power plant should be applied to all radioactive sources that could potentially harm plant personnel or the public, or contaminate the environment, taking into account a graded approach (see 3.1). The following are examples of sources that should be considered: ….	Para 3.36 completes para. 3.1, this should be indicated		No need to make reference to para introducing DiD since it is in the same section. Text modified as: 3.40 The defence in depth concept should be applied for all sources of radiation present in the nuclear power plant. The following are examples of sources of radiation likely to be present in a nuclear power plant:…		
213.	Italy	17	3.36	9		Comment: the difference between "reactor coolant system" and "reactor cooling system" is not clear	X			Considered during technical edition in accordance with IAEA Safety Glossary
214.	India	15	3.36	1st bullet	Reactor Core (including relocated damaged core)	During accident progression the core can disassemble and relocate to other potions where it needs to be confined.				This would only make the text unnecessarily

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										complicated. The source of radiation is the "fuel in the reactor core" if you prefer. When the core has been relocated, we are already in the level 4 of DiD. We don't consider a relocated core as a source of radiation to which we apply several levels of DiD.
215.	Italy	18	3.37	1	Defence in depth should be implemented with appropriate account taken of the graded approach and the fact that many radioactive sources do not qualify for all levels of defence in depth.	Typo in "graded" and "radioactive"	X			Considered during technical edition
216.	France	38	3.37		Consider deletion	This oversimplified view of graded approach is not consistent with quite complete guidance related to this topic				The paragraph addresses the radioactive sources other than the nuclear fuel, for which the implementation of DiD needs to be adapted.
217.	Indonesia	8	3.37	1,2,5 &8	~~grad-ed~~ graded ~~radio-active~~ radioactive ~~products~~ product ~~de-pending~~ depending	Replace the crossed-out words with the highlighted ones.	X			Considered during technical edition
218.	UK	32	3.37		Change to 'graded' and 'radioactive' – no hypens.	Minor typographical	X			Considered during technical edition
219.	Italy	19	3.37	8	[…] depending on the radioactive […]	Typo in "depending"	X			Considered during technical edition
220.	France	39	3.38		For each identified source of radiation, the physical barriers (including the boundaries) should be identified and ~~an evaluation~~ evaluated ~~of their robustness should be provided~~. The ~~following~~ aspects mentioned in SSG-2 should be taken into account in the evaluation. (a) to (g) shall be deleted	Robustness is not defined Added value is not clear as they do not provide guidance and use different wording as requirements, thus are not consistent with requirements.		Text modified as: 3.42 The physical barriers included in the design are an important consideration when assessing the adequacy of the implementation of defence in depth. For each identified source of radiation, the physical barriers (including the reactor coolant pressure boundary and the containment boundary) should be identified and their robustness should be evaluated in accordance with a graded approach.		The term "robustness" is used in other safety guides related to design of SSC (SSG-53 and SSG-56) and safety assessment requirements GSR Part 4 (Rev. 1). In SSR-2/1 (Rev. 1) the term used is "robust design". All of them presented in those

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										IAEA safety standards are with the same meaning that intended in this paragraph.
221.	Germany	29	3.38		For each identified source of radiation, the physical barriers (including the boundaries) should be identified and an evaluation of their robustness should be provided taking into account a graded approach. The following aspects should be taken into account in the evaluation: …..	Clarification		3.41 For sources of radiation other than the reactor core and the nuclear fuel, defence in depth should be implemented in accordance with a graded approach, with account taken of the fact that all five levels of defence in depth will not be appropriate for many sources of radiation within the plant….		
222.	UK	35	3.38 (b)		"Codes used for the design and manufacturing or construction of barriers should be appropriate. If proven materials and technologies for the manufacturing or construction are not being proposed, appropriate justification and substantiation should be provided."	Is it always appropriate to say "….proven materials and technologies for the manufacturing or construction should be used"? This seems to be standing in the way of innovation. It is appropriate to take into account the novelty or maturity of the materials and technologies in the evaluation, but that is not what the current text says. In the context of defence in depth, some innovation at some levels is easier to accept if it is backed by other levels following a more mature approach.	X	3.42 … (b) Appropriate codes and standards should be used for the design and manufacture or construction of barriers, and proven materials and technologies should be used in the manufacture or construction.		
223.	UK	36	3.38 (b)		Change to "Codes and standards used for ……"	For completeness	X			
224.	WNA	7	3.38 (c)		All loads and combination of loads that can apply to the barriers in operational states and accident conditions, including loads caused by the effects of the internal hazards and external hazards considered in the design, should be identified, calculated and be less than the applicable limits. DS514 provides recommendations for qualification of items important to safety and reference [20] of DS514 (TECDOC 1818) provides guidance regarding the assessment of equipment capability to perform reliably under severe accident conditions(footnote n°). ~~The best estimate of equipment survivability and functionality is appropriate for assessing severe accident performance.~~ For robustness, the limits should be met with adequate margins to cover uncertainties in the calculation and to avoid a cliff edge effect when loads considered for the design are slightly exceeded.	Reference should be made to formal qualification for equipment having to perform reliably under severe accident conditions, by including reference to DS514 (DS514 is at step 12 but the corresponding SSG number is not yet defined) and reference to TECDOC 1818 (which is also reference [20] in DS514 and provides guidance regarding the assessment of equipment capability to perform reliably under severe accident conditions). Still, it should be noted that "survivability" is associated with "**reasonable** level of confidence". If the concerned equipment is part of the demonstration of practical elimination, **high** level of confidence should be sought (i.e. formal qualification). Indeed, the sentence proposed to be deleted in 3.38 (c) (*~~The best estimate of equipment survivability and functionality is appropriate for assessing severe accident performance~~*.) is actually not a recommendation but a statement which should rather be included as a footnote because it does not reflect a consensual position among Member States (e.g. in	X	Text deleted		

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					Footnote : It is current practice in some Members States to consider that the best estimate of equipment survivability and functionality may be appropriate for assessing severe accident performance.	their report on 'Safety of new NPP designs', WENRA require "adequate qualification).				
225.	UK	33	3.38 (d)		Suggest (d) is deleted.	Items (c) and (d) seem to be making a distinction between 'barriers' for accident conditions and barriers for preventing early/large releases. For the first of these (c) there should be an adequate margin as discussed. For (d) it also states that there should be an adequate margin, so what is the difference between the two ? By introducing the need to prevent early/large releases here, there is also then the question of how the need for margins equates to the requirement for barrier failure to be 'highly unlikely with a high degree of confidence'. Is (d) really adding anything at this point ?	X			
226.	ENISS	25	3.37/3.38		These characteristics influence the required number of levels and the strength of the barriers and items important to safety forming the line of defence of these safety levels, de-pending on the radioactive source.	The text should be better aligned to SSR-2/1 2.14 dealing with barriers and not DiD Levels for other sources. Suggest to make 3.1/3.37/3.38 consistent in using the same idea of "barrier".		Text modified as: 3.41 … These characteristics will differ for different sources of radiation and will influence the necessary number of levels of defence in depth and the strength of each level. 3.42 The physical barriers included in the design are an important consideration when assessing the adequacy of the implementation of defence in depth. For each identified source of radiation, the physical barriers (including the reactor coolant pressure boundary and the containment boundary) should be identified and their robustness should be evaluated in accordance with a graded approach.		
227.	France ENISS	40 26	3.39		the adequacy and effectiveness of every safety provision ~~provisions~~			3.43 An analysis of the various mechanisms that could challenge or degrade the performance of the safety functions should be carried out in order to assess the adequacy of the safety provisions that are implemented to prevent the occurrence of such mechanisms or to stop their progression.		
228.	UK	37	3.39		Should read "….every safety provision…."	Minor typographical	X			
229.	France ENISS	41 27	3.40		level of conservatism and safety criteria. ~~, typically~~	Seems to be the start of a missing sentence. Remove or complete the sentence.	X			
230.	India	16	3.40	Line 5	which should be characterized by a type of ~~transient~~ safety analysis, with associated set of analysis rules, level of conservatism and safety criteria, typically.	"transient" may be replaced with "safety" which is inline with IAEA SSR 2/1 "Requirement 42: Safety analysis of the plant design". Also refer SSG-2	X	3.44 … Each plant state should be characterized by a type of safety analysis, with an applicable set of analysis rules, level of conservatism and acceptance criteria….		

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231.	Italy	20	3.40	6		Comment: apparently, the sentence is not finished (continuation expected after "typically")	X			Word "typically" was deleted. Corrected during technical edition.
232.	Indonesia	9	3.40	6	The adequacy and effectiveness of safety provisions should be assessed by performing deterministic safety analyses modelling the plant response to a given initiating event for different boundary conditions representative of each plant state, operational occurrences, DBA, DEC without significant fuel degradation and DEC with core melting, which should be characterized by a type of transient analysis, with associated set of, typically, analysis rules, level of conservatism and safety criteria, ~~typically~~ . Recommendations on conducting deterministic safety analyses for the different plant states are provided in SSG-2 (Rev.1) [8]	Consider moving typically from the end of the sentence to after the word 'of'.	X	"typically" was deleted		
233.	France	42	3.41		The performance of safety provisions at each level of defence in depth is assessed through assessment of engineering aspects and deterministic analysis involving the use of validated and verified analysis codes and models to demonstrate that acceptance criteria are met ~~with sufficient margins~~	Margin regarding criteria is technically contradictory	X	3.45 The performance of safety provisions at each level of defence in depth is assessed through assessment of engineering aspects and deterministic analysis involving the use of validated and verified computer codes and models to demonstrate that acceptance criteria are met and that there are sufficient margins to avoid cliff edge effects. Further recommendations are provided in paras 5.14 5.39 of SSG-2 (Rev. 1) [9].		
234.	Canada	34	3.41		*The performance of safety provisions at each level of defence in depth is assessed through assessment of engineering aspects and deterministic analysis involving the use of validated and verified analysis codes and models to demonstrate that acceptance criteria are met ~~with sufficient margins~~.*	To accommodate DEC conditions.	X			
235.	UK	38	3.41		Add to the end of the paragraph: "(further guidance is provided in SSG-2)"	Suggested improvement	X			
236.	Canada	35	3.42		*The reliability analysis of safety provisions for different plant states, as indicated in para. 3.34, typically uses probabilistic techniques and takes into account the plant layout, and protective provisions against or qualification for the effects of hazards, and potential commonalities in the design, manufacturing, maintenance and testing between redundant and diverse equipment*	To improve the clarity	X	3.46 The reliability analysis of safety provisions for the different plant states, as indicated in para. 3.39, typically uses probabilistic techniques and takes into account the plant layout and either protective provisions against or qualification for the effects of hazards, and potential commonalities in the design, manufacture, maintenance and testing of redundant and diverse equipment.		
237.	UK	39	3.43		Change to "….integrated into a probabilistic safety assessment…."	Terminology	X			
238.	France	43	3.44		It should be verified that adequate diversity has been implemented in the design of systems fulfilling the same ~~fundamental~~ safety function	Please be consistent with existing requirements.		3.48 It should be verified that adequate diversity has been implemented in the design of systems fulfilling the same fundamental safety		

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					in different levels of DiD ~~plant states if a simultaneous failure of those systems would result in unacceptable damage to the fuel or radiological consequences~~.	Full diversity is not expected. Independency (thus diversity) is expected between level of DiD, not plant stated		function in different plant states if a common cause failure of those systems would result in unacceptable damage to the fuel or unacceptable radiological consequences.		
239.	ENISS	28	3.44		It should be verified that sufficient diversity has been implemented in the design of systems fulfilling the same fundamental safety function in different plant states so that the ~~if a~~ simultaneous failure of those systems would not result in unacceptable ~~damage to the fuel or radiological~~ consequences.	This seems to be much more demanding than SSR-2/1 req 24 on CCF only requiring to "achieve necessary reliability".				
240.	France	44	3.45		Consider deletion	This article is not relevant at all: - Assessment is not only frequency assessment - Frequency should not be used without "estimated" in such a context - DBA is not only to to failure of AOO control		3.49 The reliability of structures, systems and components for controlling anticipated operational occurrences should be such that they are capable of reducing the number of challenges to safety systems and of contributing to preventing the occurrence of design extension conditions.		Original paragraph was modified to consider the comment. New text is proposed.
241.	WNA	8	3.45		only evolve into DBA conditions with a low frequency, ~~well~~ below the highest frequency of postulated initiating events categorized as DBAs.	"Below" is enough, we just want that if an AOO degrades to DBA, the frequency of the sequence is in the DBA range (and not in AOO range).				
242.	Canada	36	3.45		*Equipment for controlling anticipated operational occurrences is aimed at reducing the number of challenges to safety systems and thus contributes to reduce the chances that DEC involving substantial damage to the reactor will occur. ~~It should be demonstrated that their reliability is such that anticipated operational occurrences only evolve into DBA conditions with a low frequency, below the frequency of postulated initiating events categorized as DBAs.~~*	Logic of text does not seem to be correct.				
243.	Germany	30	3.45		Equipment for controlling anticipated operational occurrences is aimed at reducing the number of challenges to safety systems. It should be demonstrated that their reliability is such that anticipated operational occurrences only evolve into DBA conditions with a low frequency, well below the highest frequency of postulated initiating events categorized as DBAs.	We agree with this statement and would like to pay your attention that definition of "safety system" in IAEA Safety Glossary should be updated in accordance with it. IAEA Safety Glossary definition of a "safety system": "A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the reactor core, or to limit the consequences of design basis accidents and some anticipated operational occurrences ~~and design basis accidents~~." The assignment of "safety systems" to DBAs (DiD level 3), as presented here (para 3.45 of DS508),		Agree to take into consideration the proposed modification for future update of IAEA Safety Glossary. Agree to ensure consistency in further IAEA documents. Para was modified as: 3.49 The reliability of structures, systems and components for controlling anticipated operational occurrences should be such that they are capable of reducing the number of challenges to safety systems and of contributing to preventing the occurrence of design extension conditions.		

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
						should be consistently applied in further IAEA documents as well.				
244.	ENISS	29	3.45		~~It should be demonstrated that their reliability is such that anticipated operational occurrences only evolve into DBA conditions with a low frequency, well below the highest frequency of postulated initiating events categorized as DBAs.~~ It should be demonstrated that the reliability of equipment to manage AOO is such that in case of their failure, if the resulting event evolve into a DBA, the frequency of this event is well below the highest frequency of postulated initiating events categorized as DBAs, and the safety systems to manage such a situation are available.	The idea of SF-1 is to say if DiD level 2 fails, the next level has to be available: SSR-2/1 2.13: "If one level of protection or barrier were to fail, the subsequent level or barrier would be available...." Therefore, the point is not only a question of event frequency in the n+1 DiD level, but it should be required that the situation is manageable in this level or by one of the subsequent level or just sufficiently unlikely. Suggestion is to capture the initial idea in case of a progression towards a DBA, but to add the need to control the situation.		3.49 The reliability of structures, systems and components for controlling anticipated operational occurrences should be such that they are capable of reducing the number of challenges to safety systems and of contributing to preventing the occurrence of design extension conditions.		
245.	France	45	3.46		The ~~combined~~ reliability of the safety systems designed to ~~mitigate~~ limit the consequences of a DBA should be sufficient so that ~~to demonstrate with high confidence~~, that their probability of failure, including under the conditions expected for each accident sequence postulated, is ~~very~~ sufficiently low. ~~A failure probability below than 10-3 in order of magnitude would be consistent with the strict requirements for reliability imposed to safety systems and supported by operational experience and testing.~~ A failure probability below 10-3 failures per demand in order of magnitude would be consistent with the strict requirements for reliability imposed to safety systems and supported by operational experience and testing.	Please explain "combined reliability" Reliability of a system is not only reliability under certain conditions This article is tricky and could limit reliability analysis to prob calculation. Probability concept is not just a figure. The concepts of "very" or "high confidence" are not understandable in this context The figure is not justified. At a maximum, it could be presented as a practice in some MS For clarity		Original text in paragraph was deleted. New text is proposed: 3.50 The reliability of safety systems should be such that the collective contribution to the core damage frequency of failing to control design basis accidents does not exceed the safety goals of the plant (e.g. for new nuclear power plants typically below 10-5 per reactor-year). Design extension conditions without significant fuel degradation should be postulated for specific low frequency event sequences as appropriate to achieve the safety goals.		
246.	WNA	9	3.46		A failure probability below 10-3 in order of magnitude, for each individual safety system, would be consistent with the strict requirements	It should not be understood as a combined frequency of failure of all safety systems.				
247.	USA	1	3.46	4	~~A failure probability below 10-3 in order of magnitude would be consistent with the strict requirements for reliability imposed to s~~Safety systems and component reliability should be	Suggest not including a specific reliability target value. The level of reliability would not be necessary for very low frequency initiating events. Engineering analysis with testing and risk principles should also be considered in determining system and				

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					supported by applicable operational experience, analysis, and testing.	component reliability especially for first-of-a-kind engineered components.				
248.	ENISS	30	3.46		A failure probability below 10-3 failures per demand in order of magnitude would be consistent with the strict requirements for reliability imposed to safety systems and supported by operational experience and testing.	For clarity				
249.	Canada	37	3.46		*The combined reliability of the safety systems designed to mitigate the consequences of a DBA should be sufficient to demonstrate with high confidence, that their probability of failure ~~under the conditions expected for each accident sequence postulated~~ <u>to respond to a DBA</u> is very low.*	To improve clarity				
250.	Canada	78	3.46 **Major comment**		Delete paragraph 3.46. **It is not acceptable for a Safety Guide to add additional requirements to the Safety Standard**.	DEC-A is not a plant state. This text implies that it is. We oppose this approach where the designer classifies postulated accident scenarios based on the analysis <u>outcome</u>. The text reduces to "*DEC without significant fuel degradation shall not lead to significant fuel degradation*". The reasoning is circular and the requirement is meaningless. SSR-2/1 does not set separate requirements for DEC-A and DEC-B.	X	Original paragraph deleted. 3.17 To meet the requirements presented in paras 3.15 and 3.16, two separate categories of design extension conditions should be identified: design extension conditions without significant fuel degradation and design extension conditions with core melting. 3.18 A process for the comprehensive identification of design extension conditions without significant fuel degradation should be developed. Paragraphs 3.39 to 3.44 of SSG-2 (Rev. 1) [9] provide recommendations …		
251.	France	46	3.47		Consider deletion.	Overdemanding recommendation : it is expected to postulate systematically the failure of all safety systems during DBA		3.52 The reliability of safety features for design extension conditions without significant fuel degradation should be such that it can be demonstrated, with a sufficient level of confidence and considering applicable analysis rules (see paras 7.45-7.55 of SSG-2 (Rev. 1) [9]), that they are capable of preventing core damage with a frequency higher than the established probabilistic targets.		Not accepted, since it is intended for identifying potential DEC-B situations. New text is proposed.
252.	WNA	10	3.47		Usually, for each combination analysed, if the consequences exceed those acceptable for DBAs <u>and may cause a core melt with unacceptable frequency,</u> separate, independent and diverse safety features …	There should be a risk informedlimit in the range of DEC-A: all safety systems do not need to be diversified, only those that are used in rather high frequency AOO or DBA. It is not worth dealing with core melt sequences that have a frequency well below the core melt prevention target.		3.51 … For each such combination analysed, if the consequences exceed those acceptable for design basis accidents and might cause a core melt with unacceptable frequency, separate, independent and diverse safety features, which are unlikely to fail by the same common cause, should be implemented (e.g. an alternate AC power supply in case of a total loss of the emergency power supply, or a separate and diverse decay heat removal chain).		

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253.	Canada	79	3.48 **Major comment**	1st sentence	Change to: "*3.48 Safety features for DEC ~~without significant fuel degradation~~ should be demonstrated to be ~~efficient enough according to the applicable analysis rules to prevent core damage for the accident sequences for which they are intended and~~ sufficiently reliable in order to contribute to ensuring a core damage frequency below the established probabilistic targets. …*"	Once again, the text reduces to "*DEC without significant fuel degradation shall not lead to significant fuel degradation*". See Canada comment 1 concerning problems caused by recommending different limits for DEC-A and DEC-B.	X	3.52 The reliability of safety features for design extension conditions without significant fuel degradation should be such that it can be demonstrated, with a sufficient level of confidence and considering applicable analysis rules (see paras 7.45-7.55 of SSG-2 (Rev. 1) [9]), that they are capable of preventing core damage with a frequency higher than the established probabilistic targets.		
254.	India	17	3.49	Line 2	~~However, since the analysis of core melt and its impact on containment integrity is surrounded by considerable uncertainties~~ The safety analyses for this purpose should take due account of the uncertainties associated with analysis of core melt and its impact on the containment integrity including the aspects of structural integrity as well as heat removal capability from the containment. The capacity and reliability claimed for these safety features should be considered cautiously in consideration of these uncertainties.	Re-drafting suggested for better clarity		3.53 The capability and reliability of safety features for design extension conditions with core melting should be sufficient to ensure that the integrity of the containment will not be jeopardized during any postulated core melt sequence. However, since the analysis of core melt and its impact on the integrity of the containment is associated with considerable uncertainties, the reliability claimed for such safety features should be considered with caution.		The intention is not to describe the sources of uncertainty, which are many and not only the points that you indicated. The intention is not to provide recommendations on uncertainty analysis.
255.	Indonesia	10	3.49	6	The capacity and reliability of safety features specifically designed to mitigate the consequences of DEC with core melting should be adequate to ensure that the containment integrity will not be jeopardized during any postulated core melt sequence. However, since the analysis of core melt and its impact on containment integrity is surrounded by considerable uncertainties the reliability claimed for these safety features should be considered cautiously in consideration of these uncertainties. ~~.~~	Consider crossing-out the one dot too many after the word 'uncertainties.	X			
256.	Japan	10	3.49		The ~~capacity~~ capability and reliability of safety features specifically designed to mitigate the consequences of DEC with core melting should be adequate to ensure that the containment integrity will not be jeopardized during any postulated core melt sequence. However, since the analysis of core melt and its impact on containment integrity is surrounded by considerable uncertainties the reliability claimed for these safety features should be considered cautiously in consideration of these uncertainties.	To keep a consistency with para 3.27 and SSR-2/1 (Rev. 1).	X	3.53 The capability and reliability of safety features for design extension conditions with core melting should be sufficient to ensure…		

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257.	Italy	21	3.49	6	[…] of these uncertainties.	Typo (2 full stops)	X			Considered during technical edition
258.	France	47	3.50		The assessment should include an evaluation of the adequacy and effectiveness of the different accident management strategies defined to cope with severe accident scenarios. ~~This evaluation should demonstrate that the likelihood of an accident having unacceptable consequences for people and the environment, and which relies on both fixed and non-permanent equipment to mitigate the consequences of such an accident, is extremely low.~~	This sentence is not acceptable and contradictory with existing requirement of SSR-2/1 depending on the meaning of "unacceptable". For example, early releases is unacceptable release and use of non permanent equipment is obviously not adequate in such a case.	X	Text deleted		
259.	Canada	38	3.50			Clause is similar to safety goal	X	Text deleted		
260.	Canada	39	3.51			Reliability of support systems should be taken into account in the reliability of the safety systems.			X	The text already addresses your comment. 3.54 It should be demonstrated that the reliability of safety systems and safety features for design extension conditions is not limited by the reliability of their supporting systems.
261.	Canada	40	Section on Independence between Levels of Defence in Depth **Major comment**		Given the size and complexity of the document, Canada recommends that paragraphs 3.52 to 3.68 are deleted. Some material relevant to DEC and PE could be moved to other parts of the document if it is not there already.	This document has a main purpose of explaining DEC and PE, not discuss independence between levels of DiD. Discussion of independence "to the extent practicable" should not be hidden in a document that is primarily about DEC and PE.				The consideration of DEC and PE in the frame of DiD was agreed and presented in the DPP. The WG of NUSSC has delimited the areas to be addresses in this safety guide. This comment contradicts these agreements.
262.	France ENISS	48 31	3.56		3rd sentence: The design of a nuclear power plant should consider all potential causes of dependencies ~~and include~~ and implement an approach to remove them to the extent reasonably practicable.	To remove unnecessary wording		3.58 …The design of a nuclear power plant should consider all potential causes of dependencies and an approach should be implemented to remove them to the extent reasonably practicable.		
263.	Germany	31	3.56	Line 9	... For this reason, safety features specifically designed to mitigate the consequences of accidents with degradation or melting of the core should, as far as practicable, be independent from safety systems, in accordance with paras 4.13A and 5.29 of SSR-2/1 (Rev. 1)	Typo.	X	3.58 Because of these factors, full independence of the levels of defence in depth cannot be achieved. The design of a nuclear power plant should consider all potential causes of dependencies and an approach should be implemented to remove them to the extent		

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					[1], and also from systems used in normal operation and to mitigate AOO.-			reasonably practicable. Robust independence should be implemented among systems whose simultaneous failure would result in conditions having harmful effects for people or the environment. 3.59 As far as practicable, the sharing of systems or parts of them for executing functions for different plant states should be avoided. However, since this might not be always practical or possible, it should be ensured that within the event sequence that might follow a postulated initiating event, a system credited to respond in a given plant state will not have been needed for a preceding plant state. As emphasized in para. 4.13A of SSR/2-1 (Rev. 1) [1], this is especially important when safety systems are to be credited for the mitigation of design extension conditions (see para. 3.65).		
264.	Japan	11	3.56	Last sentence	For this reason, in paras 4.13A and 5.29 of SSR-2/1 (Rev. 1) [1] states that in particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) ~~specifically designed to mitigate the consequences of accidents with degradation or melting of the core should~~ shall~~,~~ as far practicable~~,~~ be independent ~~from~~ of safety systems~~, in accordance with paras 4.13A and 5.29 of SSR-2/1 (Rev. 1) [1], and also from systems used in normal operation and to mitigate AOO~~.	This is not a recommendation just using "should sentence" the requirement. Should be just referred the requirement as it is.				
265.	Germany	32	3.57		It is necessary to demonstrate that the effectiveness of the levels of defence is not reduced by factors that compromise the independence of the levels of defence in depth. These factors include the following ~~are as follows~~: (a) The sharing of systems or parts of systems for executing functions for different plant states, for example for normal operation and for design basis accidents. (b) Common cause failures that can impact different levels of defence in depth. Typical root causes of such failures are undetected human errors in design or manufacturing, human errors in the operation or maintenance, inadequate equipment qualification or inadequate protection against internal or external hazards.	Clarification		3.57 The potential for common cause failures is a second factor that can compromise the independence of the levels of defence in depth. Typical root causes of common cause failures are undetected human errors in design or manufacturing, human errors in the operation or maintenance, inadequate equipment qualification or inadequate protection against internal or external hazards. Requirement 24 of in SSR-2/1 (Rev. 1) [1] states:		
266.	France ENISS	49 32	3.59		As far as practicable, the sharing of systems or parts of them for executing functions for different categories of plant states should be ~~prevented~~ avoided.		X			
267.	ENISS	33	3.59		As far as practicable, the sharing of systems or parts of them for executing functions for different categories of plant states should be prevented. However, since this might not be always practical or possible, it should be ensured that within the sequence of events that may follow a postulated initiating event, a system credited to respond in a given plant	The part after "However" applied to the containment is not clear. For example, it is saying that the containment isolation system used for a DBA, can't be used if the situation is degrading to a DEC without core melt. It is not thought this is the intent here. The point is more to ensure that this system is still available.		3.59 As far as practicable, the sharing of systems or parts of them for executing functions for different plant states should be avoided. However, since this might not be always practical or possible, it should be ensured that within the event sequence that might follow a postulated initiating event, a system credited to respond in a given plant state will not have been needed for a preceding plant state. As		

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					condition should not have been ~~needed~~ impaired in ~~for~~ a preceding condition			emphasized in para. 4.13A of SSR/2-1 (Rev. 1) [1], this is especially important when safety systems are to be credited for the mitigation of design extension conditions (see para. 3.65).		
268.	Canada	80	3.59	First sentence	The meaning of "safety division" should be explained (either here or by reference).	"Safety division" is not a term used in the IAEA Safety Glossary, SSR-2/1, SSR-2/2 or SSG-2.				Text deleted
269.	UK	40	3.59		Delete the last two sentences, or re-word.	The last two sentences are unclear. The first talks about 'safety systems' credited for the mitigation of DEC, so presumably these have not failed earlier in the sequence. The next refers to complementary 'safety features' to mitigate DECs, which presumably are different to the aforementioned safety systems. If so, what point is being made ?	X			
270.	WNA	11	3.60		The SSCs needed for each postulated initiating event should be identified, and it should be shown by means of engineering analyses that the SSCs needed for implementing any one defence in depth level are sufficiently independent from the other levels. One should keep in mind that a PIE is generally a bounding event covering different kinds of initiating failures and it may be difficult to list exactly all the normal operation equipment that may be initially affected by the PIE in a given DBA or DEC accident sequence. For this reason, the credit of normal operation systems in the safety assessment of DBA or DEC should be considered with extreme caution and would therefore generally not be recommended. The adequacy of…	Use of normal operation systems, especially in DEC sequences that are often meant to cover a very broad range of initial failures, is very questionable from an independence point of view. An explicit demonstration that the normal operation system remains available and operates as expected, is difficult to establish, in particular if the accident sequence is supposed to cover possible consequences of internal hazards.		3.62 Engineering assessment, deterministic and probabilistic methods should be used to assess the independence of the levels of defence in depth. The structures, systems and components needed for each postulated initiating event should be identified, and it should be shown by means of engineering analyses that the structures, systems and components needed for implementing each level of defence in depth are sufficiently independent from those for the other levels. A postulating initiating event is generally a bounding event covering different kinds of initiating failure and so it might be difficult to list all equipment for normal operation that might initially be affected by the postulated initiating event for particular design extension conditions. For this reason, the crediting of systems for normal operation in the safety assessment of design extension conditions should be considered with extreme caution and should be adequately justified. The adequacy of the independence that is achieved for each level of defence in depth should also be assessed by probabilistic analyses.		
271.	France ENISS	50 34	3.61		As per SSR-2/1 req 21 and 24, the redundant, or diversified ~~The systems~~ components, as well as ~~and~~ the components required to be protected against a common cause failure to ensure the independence between ~~used for~~ different plant states, should be separated~~, within the same safety division, from one another~~ by distance or protective structures~~. if there is a possibility for consequential failures arising from a failure of a system or component of one safety division in the same safety division for another plant state.~~	The term "safety division" is not defined. The wording "The systems and components used for different plant states" is very general and large making it difficult to understand what really needs to be separated from what. Do you mean a component used in DBA1 and a component used in DBA2 have to be separated? The components in the containment are typically used in different plant states, but what has to be separated from what in that case?		Text deleted and Req. 21 and 24 of SSR-2/1 (Rev.1) were recalled.		

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						Justification of proposed wording: to separate A and B, the A and the B have to be identified. The proposition Referring to "redundant and diversified components" intend to identify A and B, as well as the following sentence, trying to make the link with the previous text. Hope that captures the intended idea.				
272.	WNA	12	3.61		The systems and components used for different plant states should be separated, within the same safety division, from one another by distance or protective structures if there is a possibility for consequential failures arising from a failure of a system or component of one safety division in the same safety division for another plant state and if it leads to the complete loss of all means to control a safety function in a situation where it should be required.	The recommendation seems clever but it is almost impossible to fulfil with reasonable lay-out provisions. The sufficiency of separation should not be assessed division by division but as a whole, at the plant level		Text deleted and Req. 21 and 24 of SSR-2/1 (Rev.1) were recalled.		
273.	Canada	81	3.61	First sentence	Change first sentence to, "*3.61 The systems intended for mitigating severe accidents should be functionally and physically separated from the systems intended for other plant states to the extent practicable*."	Independence is only required "to the extent practicable" by SSR-2/1 § 5.29 (a)..	X	3.61 The systems intended for mitigating severe accidents should be functionally and physically separated from the systems intended for other plant states to the extent practicable….		
274.	UK	41	3.62		Suggested wording: "For most reactor designs, the reactor trip system is designed as a safety system for accident conditions that is also needed for the control of AOOs."	With regard to "For most reactor designs, the reactor trip system is designed as a safety system that is also needed for the control of accidents." From the 2018 safety glossary, AOOs are operational states, whilst DBA and DECs are accident conditions. A safety system is provided to ensure shutdown, heat removal or limit the consequences of an AOO or DBA.	X			
275.	France	51	3.64		For instrumentation and control systems, it should be demonstrated that ~~defence in depth within the overall instrumentation and control architecture is achieved by means of independent lines of defence, so that the failure of one line of defence is compensated for by the following one. This can be~~ adequate independency is achieved (see notably requirement 64 of SSR-2/1 rev.1) ~~by implementing independence between different levels of defence in depth and independence between redundant functions and by design for reliability. Means of supporting design for reliability and reducing the likelihood of common cause failures in I&C systems are physical separation, electrical isolation, functional independence and independence~~	This topic is very tricky and a reference to existing requirement and guidance is sufficient This oversimplified article challenges consistency with existing standards.		Text deleted and Req. 64 of SSR-2/1 (Rev.1) was recalled.		

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					~~from the effects of communications errors, and diversity~~. Further recommendations are provided in SSG-39 [7].					
276.	Japan	12	3.64		For instrumentation and control systems, it should be demonstrated that defence in depth within the overall instrumentation and control architecture is achieved by means of independent ~~lines~~ levels of defence, so that the failure of one ~~line~~ level of defence is compensated for by the following one. This can be achieved by implementing independence between different levels of defence in depth and independence between redundant functions and by design for reliability.	Clarification. “lines of defence” is only used in SSG-39 para 4.10, but generally it is not used in SSR-2/1 (Rev. 1) as well as the safety glossary. So “lines” should be replaced by “levels”.		Text deleted and Req. 64 of SSR-2/1 (Rev.1) was recalled.		
277.	Japan	13	3.64	Last sentence	Further recommendations are provided in para. 6.38 in SSG-39 [7].	It should be specified the para. number for the referred safety guides.			X	Preference was given to quote the requirements instead of making reference to SSG-39
278.	Indonesia	11	3.65	6-7	The assessment of the implementation of defence in depth should demonstrate that independence between successive levels of defence is adequate to limit the progression of deviations from normal operation and to prevent harmful effects to the public and the environment should accidents occur. For this purpose, the assessment of the implementation of the defence in depth should aim to verify that the vulnerabilities for common cause failures, originated in the layout, design, manufacturing, operation and maintenance, between ~~structures, systems and components~~ SSCs that are claimed to be independent, have been identified and removed to the extent practicable. In particular, functional dependencies should be removed or justified.	Replace structures, systems and components with SSCs to be consistent with previous style of writing	X			
279.	France ENISS	52 35	3.66		The assessment should demonstrate that the operability of the safety systems is not jeopardized by failures in systems designed for normal operation ~~or anticipated operational occurrences.~~ Following an event, the failures occurring in anticipated operational occurrences should not compromise the capability of safety systems to manage the event if escalating to a DBA.	By definition (safety glossary), the systems designed to limit the consequences of AOO are safety systems. The sentence is therefore saying: safety systems not jeopardised by failures in safety systems for AOO, what is not achievable.		3.64 The assessment should demonstrate that safety systems that are intended to respond first in an accident are not jeopardized by the initiating event. The assessment should demonstrate that the operability of the safety systems is not jeopardized by failures in systems designed for normal operation. Following an initiating event, the failures occurring in anticipated operational occurrences should not compromise the capability of safety systems to manage a design basis accident.		
280.	Japan	14	3.66		The assessment should demonstrate that the safety ~~features~~ systems intended to respond first are not jeopardized by the initiating event. The	Correction.	X			

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					assessment should demonstrate that the operability of the safety systems is not jeopardized by failures in systems designed for normal operation or anticipated operational occurrences.					
281.	Egypt	4	3.67		For this purpose, the assessment should provide evidence that the reliability, redundancy, diversity and independence of the support service are ~~is~~ commensurate with the significance to the safety of the system being supported.	Grammar	X			
282.	ENISS	36	3.68		The demonstration of sufficient independence between lines of ~~An assessment of independence of SSCs that may be necessary at~~ different levels of defence in depth should not be jeopardized neither by the occurrence of an ~~to mitigate the consequences of a single or a likely combination of~~ external hazard nor by the occurrence of a likely combination of external hazards, where such an event may lead to a common cause failure. In that perspective, despite the hazard event and any consequential failures (including any potential common cause failures) ~~on the plant should be conducted.~~ it should be demonstrated that the available SSCs are sufficient to mitigate the consequences and ensure that radiological consequences are kept below acceptable limits for DBA. ~~postulated initiating event and the failures induced in the plant cannot result in common cause failure between the SSCs necessary for its mitigation at different levels of defence in depth.~~ In addition, the DEC with core melting ~~In particular, the necessary~~ safety features ~~for design extension conditions for core melting~~ should ~~always remain available.~~ be adequately protected against external hazards to avoid situations where DBA safety systems and DEC with core melting safety features are affected at the same time. This may be an appropriate geographical separation to avoid a common cause failure originated by a hazard event.	The text is a bit complicated to understand and to apply to hazards such as external flooding, lightning, extreme climatic conditions, external fires, industrial hazards, airplane crash. The last sentence "always remain available" is much more demanding than SSR-2/1 and more or less requiring that the DEC with core melting features are in a separate bunker protected against airplane crash and available whatever hazards the plant may see. What about the necessary periodic tests where the features are unavailable? As the last line of defence, DEC with core melting safety features may be protected, but this should be tempered a bit and kept as something practicable. The failure of all DEC with core melting features (Airplane crash) is acceptable if all safety systems are available. Suggestion in the proposed text is to emphasise the need to avoid CCF caused by hazards and to protect DEC with core melt safety features where necessary.	X	3.66 An assessment should be conducted of the independence of structures, systems and components that might be necessary at different levels of defence in depth to mitigate the consequences of a single hazard or a likely combination of internal or external hazards on the plant. It should be demonstrated that the postulated initiating event and the failures induced in the plant cannot result in common cause failure of the structures, systems and components necessary for mitigation of the hazard at different levels of defence in depth. In particular, the necessary safety features for design extension conditions for core melting should always remain available.		
283.	Germany	33	3.68		An assessment of independence of SSCs that may be necessary at different levels of defence in depth to mitigate the consequences of a single or a likely combination of external or internal hazards on the plant should be	Clarification	X	3.66 An assessment should be conducted of the independence of structures, systems and components that might be necessary at different levels of defence in depth to mitigate the consequences of a single hazard or a likely		

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					conducted. It should be demonstrated that the postulated initiating event and the failures induced in the plant cannot result in common cause failure between the SSCs necessary for its mitigation at different levels of defence in depth. In particular, the necessary safety features for design extension conditions with ~~for~~ core melting should always remain available.			combination of internal or external hazards on the plant. It should be demonstrated that the postulated initiating event and the failures induced in the plant cannot result in common cause failure of the structures, systems and components necessary for mitigation of the hazard at different levels of defence in depth. In particular, the necessary safety features for design extension conditions for core melting should always remain available.		
284.	Japan	15	3.68		An assessment of independence of SSCs that may be necessary at different levels of defence in depth to mitigate the consequences of a single or a likely combination of internal or external hazards on the plant should be conducted. It should be demonstrated that the postulated initiating event and the failures induced in the plant cannot result in common cause failure between the SSCs necessary for its mitigation at different levels of defence in depth. In particular, the necessary safety features for design extension conditions for core melting should always remain available.	Internal hazards also should be taken into account.	X			
285.	India	18	3.68		*Suggestion* This being an important clause, further clarity w.r.t consideration of human induced external hazards will be useful in applying of the clause for assessment.	To bring better clarity	X			
286.	Canada	82	Un-numbered paragraph after 4.1	last sentence.	No change required.	We agree with this way of dealing with the inconsistency in SSR-2/1. "*This guide refers to early radioactive releases in relation to the practical elimination of the conditions leading to them.*"	X			
287.	USA	2	Proposed Section 4.xx		A summary-level report should be provided for an NPP design that addresses the elements of DS508. The report would illustrate that any additional design alternatives for risk reduction were systematically evaluated and determined to not be necessary.	It is unclear what written product is anticipated to communicate the results for demonstration of practical elimination. Demonstration of practical elimination is not only for the design-vendors and regulators, but also for communicating to other stakeholders including the public.	X	Covered in 4.43 as 4.43 The safety analysis report of the plant should reflect the measures taken to demonstrate the practical elimination of event sequences that could lead to an early radioactive release or a large radioactive release. The safety analysis report should include, either directly or by reference, all elements of the demonstration, including the approach used to identify such event sequences, the design and operational safety provisions implemented to ensure that the possibility of such event sequences arising has been practically eliminated and the corresponding analyses.		

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288.	USA	3	Proposed Section 4.xx		In addition to the safety analysis using the DSA, PSA, and evaluations severe accident phenomena, there should be an assessment of the candidate design alternatives to prevent or mitigate the consequences of a severe accident (DEC with core damage) that were not incorporated in the final design because additional risk reduction was unwarranted. A cost-benefit assessment of these severe accident design alternatives may be performed to demonstrate that remaining residual risk of the new NPP design does not justify incorporation of additional candidate severe accident design features.	The approach currently presented in DS508 is certainly one way of achieving the objective of demonstrating, in part, practical elimination has been achieved. The USNRC requires that all NPP design-vendors evaluate and possibly include additional severe accident prevention and mitigation of the consequences of a severe accident. In addition to DSA, PSA, severe accident phenomena and mitigation, containment performance being analyzed by the design-vendor and reviewed by the regulator, USNRC has had a long-standing requirement that new reactor designs satisfy environmental requirements in particular for severe accidents. This requirement includes evaluating severe accident mitigation design alternatives (SAMDA) and documenting the evaluations in environmental reports. A cost-benefit approach using PSA results is applied to the design using the SAMDA process for the given NPP site parameters. This environmental assessment is finalized at the end of the design-certification process after DSA, PSA, and beyond design basis events (DEC-A and DEC-B) have been evaluated for safety of the NPP design. The SAMDA/environmental assessment demonstrates that other candidate design alternatives were examined systematically. The determination of this environmental assessment is that there will be no significant offsite impact to the public.			X	Covered in section 4.
289.	Canada	83	Section 4	general	Number the unnumbered paragraphs 4.1 and 4.7. Verify the cross-references in section 4.	There are problems with numbering in section 4. When they are corrected, the problems with cross-references later in the section may be resolved, but should be verified.	X			
290.	Italy	22	4			Formatting is different from this point on	X			Considered during technical edition
291.	Egypt	5	4.1		…also introduces the expectation that event sequences that would lead to an early radioactive release or a large radioactive release….	Editorial issue	X			Considered during technical edition
292.	UK	42	4.1		Erroneous quotation marks	Minor typographical	X			
293.	Canada	84	4.2	Editorial	Change ‘i.e.’ to ‘e.g.’. “4.2 With regard to design, ‘practical elimination’ is normally considered to refer only to those events or sequences of events	Should be “e.g.” not “i.e.” • e.g means “for example” (Latin *exempli gratia*)	X			

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					leading to or involving significant fuel degradation, ~~i.e.~~ e.g. a 'severe accident', for …"	• i.e. means "that is" (Latin *id est*) In this case, severe accidents are only an example of significant fuel degradation as IAEA Glossary definition of severe accident only refers to "significant core degradation" and so does not include degradation of the fuel in spent fuel storage				
294.	Indonesia	12	4.2		~~With regard to~~ regarding the design, 'practical elimination' is normally considered to refer only to those events or sequences of events leading to or involving significant fuel degradation, i.e., a 'severe accident', for which the confinement of radioactive materials cannot be reasonably achieved. Those event sequences have to be considered in the design for 'practical elimination', either by physical impossibility or by being extremely unlikely to occur with a high level of confidence.	Consider substituting regarding for with regard to, since regarding is more direct and simpler	X			Considered by the technical editor
295.	Germany	34	4.2	Line 4	… Those event sequences have to be considered in the design for 'practical elimination', either by physical impossibility (see also paras. 4.35 - 4.36) or by being extremely unlikely (see also paras. 4.37 – 4.42) to occur with a high level of confidence	References for clarification	X	4.4 The concept of practical elimination is normally applied only to those events or sequences of events that could lead to or involve significant fuel degradation, i.e. a severe accident, for which the confinement of radioactive material cannot be reasonably achieved. The practical elimination of such plant event sequences is required to be ensured by design [1], either ensuring that the plant event sequence is physically impossible (see paras 4.34–4.35) or because the plant event sequence is considered, with a high level of confidence, to be extremely unlikely to arise (see paras 4.36–4.43).		
296.	Canada	41	4.3		To reduce the apparent contradiction, we suggest rewording slightly: "*The concept of 'practical elimination' ~~should be considered as~~ is part of the overall safety approach for ~~the design of~~ nuclear power plants in accordance with Chapter 2 of SSR-2/1 …*"	It is an unfortunate choice of words to imply that 'practical elimination' is considered in the design when, according to the definitions in SSR-2/1, accidents more severe than DEC are NOT considered in the design! The only plant states considered in the design are NO, AOO, DBA and DEC. Given the comprehensive nature of the design, inspection and administrative controls for Reactor Pressure Vessel fabrication and installation, this is clearly untrue (for example, see DS508 Annex I). But this is a problem with SSR-2/1 and cannot be resolved in DS508.			X	The concept of practical elimination is considered in the design, as the objective of the recommendations in this safety guide. Para 4.5 provides this explanation. 4.5 The concept of practical elimination should be applied as part of the overall safety approach to the design of nuclear power plants, as set out in section 2 of

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										SSR-2/1 (Rev. 1) [1]. As a result of the implementation of the first, second, third and fourth levels of defence in depth, the likelihood of an off-site radioactive release that could potentially result from an accident will be very low. However, it is necessary to verify that there would not be credible plant conditions that could not be effectively mitigated and which could thus lead to unacceptable radiological consequences. This is where the aim of the practical elimination concept lies: to reinforce the implementation of defence in depth at a plant by a focused analysis of those conditions having the potential for unacceptable radiological consequences.
297.	WNA	13	4.3		Practical elimination should not be seen as an alternative to severe accident mitigation: instead, efficient and reliable provisions should be implemented to mitigate any core melt sequence, in accordance with the defence in depth concept <u>and practical elimination should only be implemented to deal with the few remaining core melt phenomena that cannot be reasonably addressed in the design</u>.	Full agreement with the recommendation ; it is worth making it even more explicit		4.6 Practical elimination should not be seen as an alternative to mitigation of the consequences of a severe accident (i.e. implementation of the fourth and fifth levels of defence in depth); rather, the application of practical elimination should be in addition to the provision of safety features for design extension conditions with core melting, and on-site and off-site emergency response facilities. Moreover, the practical elimination of event sequences that could lead to a large radioactive release or an early radioactive release does not remove the need for emergency preparedness and		
298.	Canada	42	4.3	last statement	*Practical elimination should not be seen as an alternative to severe accident mitigation: instead, efficient and reliable provisions should*	To improve clarity				

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					be implemented to mitigate any core melt ~~sequence~~ consequence, in accordance with the defence in depth concept.			response, in accordance with principle 9 of SF-1 [3] and the requirements of GSR Part 7 [12].		
299.	France ENISS	53 37	4.5		In situations of limited confinement, for example in accidents involving fuel storage or when the containment is open and cannot be closed in time, or where ~~there~~ is a ~~an~~ containment bypass that cannot be isolated, the only way to prevent unacceptable releases is to avoid the occurrence of a severe accident. In such cases, it may be necessary to demonstrate ~~practical elimination by showing~~ with a high degree of confidence that such severe accidents would be extremely unlikely.	Clarification/simplification		4.12 …In particular, in situations of limited confinement, for example in accidents involving fuel storage or when the containment is open and cannot be closed in time, or where there is a containment bypass that cannot be isolated, the only way to prevent unacceptable radiological consequences is to prevent the occurrence of such severe accidents. In such cases, it may be necessary to demonstrate practical elimination by proving the physical impossibility of the accident or by proving with a high degree of confidence that such severe accidents would be extremely unlikely.		
300.	Germany	35	4.5		When a severe accident occurs, it is necessary to ensure that radioactive materials released from the nuclear fuel will be confined. In situations of limited confinement, for example in accidents involving fuel storage or when the containment is open and cannot be closed in time, or there is an containment bypass that cannot be isolated, the only way to prevent unacceptable releases is to avoid the occurrence of a severe accident. In such cases, it may be necessary to demonstrate practical elimination by showing the "physical impossibility" of the accident or prove with a high degree of confidence that such severe accidents would be extremely unlikely.	Both ways of demonstrating practical elimination must be addressed here. The demonstration of "physical impossibility" is also relevant in situations of limited confinement.		4.12 If a severe accident occurs, it is necessary to ensure that radioactive material released from the nuclear fuel will be confined. In particular, in situations of limited confinement, for example in accidents involving fuel storage or when the containment is open and cannot be closed in time, or where there is a containment bypass that cannot be isolated, the only way to prevent unacceptable radiological consequences is to prevent the occurrence of such severe accidents. In such cases, it may be necessary to demonstrate practical elimination by proving the physical impossibility of the accident or by proving with a high degree of confidence that such severe accidents would be extremely unlikely. Therefore, the issue when considering whether a particular plant event sequence should be practically eliminated is the potential for the event sequence to lead to a failure of the confinement function.		
301.	Indonesia	13	4.5	4	When a severe accident occurs, it is necessary to ensure that radioactive materials released from the nuclear fuel will be confined. In situations of limited confinement, for example in accidents involving fuel storage or when the containment is open and cannot be closed in time, or there is ~~an~~ a containment bypass that cannot be isolated, the only way to prevent unacceptable releases is to avoid the occurrence of a severe accident. In such cases, it may be necessary to demonstrate practical elimination by showing with a high degree of confidence that such severe accidents would be extremely unlikely.	Replace an containment bypass with a containment bypass	X			

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
302.	UK	43	4.5		Change to "… a containment bypass…."	Minor typographical	X			
303.	France	54	4.6		~~SSR-2/1 (Rev. 1) [1] does not provide quantitative acceptance criteria for the radiological consequences of accident conditions, or for the magnitude of what is to be considered a large radioactive release. An early radioactive release should be defined site specific considering the time restrictions to implement off-site protective measures. Therefore, acceptable limits for radiation protection, as well as probabilistic criteria or target values for the purpose of demonstrating the low frequency of a core damage accident or accident sequences leading to radioactive releases, should be established, consistent with the regulatory requirements., consistent with the regulatory requirements.~~	This article is not acceptable. It is not technically consistent with some member states practices: - large releases definition do not need to be quantified. The corresponding situation "qualitatively" lead to unacceptable releases; - no probabilistic criterion is needed as PE relies primarily on deterministic justification At a minimum, replace it by an article that quotes "some member states practices"		4.8 SSR-2/1 (Rev. 1) [1] does not provide quantitative acceptance criteria for the radiological consequences of accident conditions, nor for the magnitude of what is to be considered an early radioactive release or a large radioactive release. In some States an early radioactive release is defined for a specific site considering restrictions on implementing off-site protective actions in a timely manner. In some States, acceptable limits on radioactive releases for purposes of radiation protection, and probabilistic criteria or target values for the purpose of demonstrating a low frequency of a core damage accident, have been established, consistent with regulatory requirements or objectives. However, the justification that a plant event sequence has been practically eliminated should rely primarily on a deterministic evaluation and should not be solely demonstrated by demonstrating compliance with such probabilistic criteria.		The proposal of "in some Member States" was considered since there is no contradiction because acceptable releases is on the responsibility of national regulatory authorities.
304.	Germany	36	4.6		… Therefore, acceptable limits for radiation protection, as well as probabilistic criteria or target values for the purpose of demonstrating the low frequency of a core damage accident or accident sequences leading to radioactive releases, should be established, consistent with the regulatory requirements. It should be noted that the 'practical elimination' cannot alone be demonstrated by showing the compliance with these probabilistic values.	This should be added here since it has an impact on the understanding of probabilistic criteria.		4.8 SSR-2/1 (Rev. 1) [1] does not provide quantitative acceptance criteria for the radiological consequences of accident conditions, nor for the magnitude of what is to be considered an early radioactive release or a large radioactive release. In some States an early radioactive release is defined for a specific site considering restrictions on implementing off-site protective actions in a timely manner. In some States, acceptable limits on radioactive releases for purposes of radiation protection, and probabilistic criteria or target values for the purpose of demonstrating a low frequency of a core damage accident, have been established, consistent with regulatory requirements or objectives. However, the justification that a plant event sequence has been practically eliminated should rely primarily on a deterministic evaluation and should not be solely demonstrated by demonstrating compliance with such probabilistic criteria.		
305.	Indonesia	14	4.6		SSR-2/1 (Rev. 1) [1] does not provide quantitative acceptance criteria for the radiological consequences of accident conditions, or for the magnitude of what is to be considered a large radioactive release. An early radioactive release should be defined site specific considering the time restrictions to implement off-site protective measures. Therefore, acceptable limits for radiation protection, as well as probabilistic criteria or target values for the purpose of demonstrating the low frequency of a core damage accident or accident sequences leading to radioactive releases, should be established, consistent with the regulatory requirements.					

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306.	UK	45	4.6		Change last sentence: "Therefore, acceptable limits for radiation protection, as well as probabilistic criteria or target values for the purpose of demonstrating the low frequency of a core damage accident or accident sequences leading to radioactive releases, should be established, consistent with ~~the~~ any regulatory requirements.	The UK has a goal setting regulatory regime and as a result ONR does not prescribe regulatory requirements to the level of specificity as LWR PSA criteria for core damage. A simple change from "the regulatory requirements" to "any regulatory requirements" would cover the UK position.				
307.	UK	44	4.6		Change to "…defined for a specific site considering…."	Improve wording				
308.	France	55	4.7		~~When~~ if defining ~~these~~ radiological criteria or targets for early and large releases, it is necessary …	See 4.6	X	Text deleted		
309.	Canada	43	4.7 **Major comment**	Sentence 1, 2	The text in DS508 must be revised to agree with SSR-2/1 or SSR-2/1 must be revised to allow the DS508 interpretation.	Canada strongly disagrees concerning the "significant difference" and "quantitative step" between the maximum acceptable releases in DEC and the magnitude of release for practical elimination. See earlier comment on para 2.8**Error! Reference source not found.**.	X	Text deleted		
310.	Canada	44	4.7 **Major comment**	Sentence 2	Delete sentence 2.	It is not true that a difference between consequence limits for DEC and PE provides a safety margin. The difference leaves a gap. In a previous response to this comment, the authors said "*If a criterion for practical elimination would be 200 T-becquerels of Cs, it is not acceptable a design that would consider a release of 199 T-becquerels, or anything closer, a 'successful mitigation'.*" This may be true. A release of perhaps 1TBq might be acceptable for DEC as "protective actions that are limited in terms of lengths of time and areas of application shall be sufficient for the protection of the public". What are the requirements for releases in the gap between 1 TBq and 200 TBq?	X	Text deleted		
311.	Canada	85	4.7 **Major comment**	last 2 sentences	Delete the last 2 sentences. "~~The concept of "practical elimination" is only applied in relation to plant conditions that can lead to early radioactive releases or large radioactive releases, for which reasonably practicable technical means for their mitigation cannot be implemented. Otherwise, such means should be considered under the strategy for accident mitigation and would not be part of the~~	This needs much more consideration. The final sentence of the paragraph implies a new SSR-2/1 requirement that scenarios that CAN be mitigated by reasonably practicable means MUST be mitigated. This is despite the fact that they are beyond the "plant states considered in the design". **It is not acceptable for a Safety Guide to add additional requirements to the Safety Standard**. It will need SSR-2/1 to be revised since PE is currently supposed to apply to all large or early	X	Text deleted		

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					~~application of the concept of practical elimination.~~"	releases. This proposal restricts PE only to those where reasonably practical means to mitigate the release cannot be implemented. I think it is an example of an approach where the PE concept is applied only to very high consequence scenarios, e.g. • releases that are very large • releases that are large AND early Presumably, this type of scenario would also need to be less frequent than DEC by a significant margin following the established logic that higher consequence events must have lower frequency of occurrence. Current SSR-2/1 5.31 does not mention this and uses the same wording for the limits to permissible consequences in DEC as are used to define large or early release. Basically, if limited offsite protective actions are effective, the scenario is allowed in the DEC frequency range. If limited offsite protective actions are not effective, the scenario must be PE. We have very similar objections to this interpretation of SSR-2/1 as we had to the earlier draft of DS508 paras 2.8 and 4.7. We like the idea, but it conflicts with the wording of SSR-2/1.				
312.	India	19	4.8	Line 3	The first step for demonstrating the practical elimination of plant conditions that can lead to an early radioactive release or a large radioactive release is the identification of severe accident sequences having the potential to give rise to 'unacceptable radioactive releases' <u>using the PSA Level-2 insights.</u>	PSA Level-2 which also involves DEC insights with containment failures can provide good insights and assurance on the practical elimination of the events.				Not accepted PSA doesn't analyse what is not postulated. The occurrence of events to be demonstrated to be practically eliminated is not approach probabilistically PSA is not used in the identification but in the assessment.
313.	France	56	4.9		The concept of practical elimination' …~~. Where further features could be implemented, either for prevention of accidents or for mitigation of the consequences, they should be considered, as far as reasonably practicable~~	This sentence implies that PE may be not feasible even if necessary	X	Accepted to delete the text proposed to be deleted. The rest of the paragraph is also deleted because it is all explained in much more detail in the section on identification and assessment of safety provision.		

TITLE: DS508 - Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
314.	Canada	45	4.9	Sentence 1	Suggest rewording slightly: *The concept of 'practical elimination' is used to demonstrate that ~~design~~ provisions have been implemented, across all levels of defence in depth to ensure that plant conditions for which a large radioactive release or an early radioactive release could not be prevented, are physically impossible or highly unlikely with a high degree of confidence.*	Similar to issue with DS508 para 4.3, the text is inconsistent with SSR-2/1 which only defines "plant states considered in design". This implies that accidents with lower frequency / higher consequence than DEC are not considered in design. We believe this is a problem with SSR-2/1. DS508 cannot fix it. Also, it is not only design provisions that are used to practically eliminate some event sequences.				
315.	Canada	46	4.10		As part of the overall safety approach, the 'practical elimination' concept should be applied to a new nuclear power plant ~~at the earliest design~~ from an early stage, when it's more practicable to ~~design and~~ implement additional safety ~~features~~ provisions. The incorporation of such ~~features~~ provisions is an iterative process using insights from ~~engineering~~ experience, and from deterministic safety analyses and probabilistic safety analyses in a complementary manner.	Once again, there is a problem with considering PE in the design when PE accidents are outside of the "plant states considered in the design" according to the SSR-2/1 definition. And again, non-design provisions are not acknowledged.	X	4.9 The concept of practical elimination should be applied in a new nuclear power plant from an early stage, when it is more practicable to design and implement additional safety features. The incorporation of such features is an iterative process, which should use insights from engineering experience, and from deterministic safety analyses and probabilistic safety assessment in a complementary manner.		
316.	France ENISS	57 38	4.12	footnote 4	Currently, the technology used for equipment hatches ~~is generally~~ may not be fast enough to ensure re-closure and restoration of the containment integrity, before significant activity release occurs. Therefore, any significant rapid fuel degradation mechanism in shutdown operating modes with an open containment should be considered for 'practical elimination'	Equipment hatch closure is claimed in a number of member states where it can be conservatively demonstrated that it can be achieved in the timescale required.	X	The footnote was modified, currently footnote 15: In many LWR designs, the technology used for equipment hatches might not be fast enough to ensure re-closure and restoration of the containment integrity before a radioactive release occurs. In addition, last sentence was deleted to avoid a recommendation in the footnote.		
317.	Canada	47	4.12,	item d) iii) Footnote 4	Delete footnote 4.	Footnote 4 makes an unjustified assumption that PE of rapid fuel damage mechanisms is the only possible solution. Fast closing hatches are clearly another. Constraining design options in this way is not appropriate.			X	Last sentence was deleted to avoid a recommendation in the footnote.
318.	UK	46	Page 23	Footnote 4	Suggest change to footnote text: "On many LWR designs, the technology used for equipment hatches is generally not fast enough to ensure re-closure and restoration of the containment integrity. Therefore, unless specific design provision is included, any significant rapid fuel degradation mechanism in shutdown operating modes with an open containment should be considered for 'practical elimination'.	The point on equipment hatches is important and relevant, but is perhaps too definitive. From ONR's experience of assessing modern LWR designs, many do not claim to be able to close equipment hatches quickly. However, some have made specific design provision for this, eg to facilitate in-vessel retention and passive recirculation.				
319.	Italy	23	4.12	b iii	carbon monoxide;	Semicolon, as the list is not finished yet	X			Considered during technical edition

TITLE: DS508 - Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
320.	France ENISS	58 39	4.12	footnote 5	Most plant designs in various States locate the spent fuel pool outside of the containment, given the slow kinetics of accidents likely to lead to severe damage of the fuel assemblies stored in the spent fuel pool. The timescales enable the implementation of on-site or off-site prevention or protective measures. ~~This option is considered as the best choice in the decision making process compared to the additional costs and operational constraints if the spent fuel pool were also located in the reactor building.~~ However, this does mean that any occurrence of significant fuel degradation in the pool would directly lead to a large radioactive release. Therefore, any accident sequence with significant degradation of the fuel assemblies stored in the spent fuel pool has to be considered for ‘practical elimination’	A guide should not comment on what is the best option unless it is irrefutable. Either option has advantages and disadvantages so both are valid.	X	Currently footnote 16: Several plant designs locate the spent fuel pool outside of the containment, given the slow kinetics of accidents likely to lead to severe damage of the fuel assemblies stored in the spent fuel pool. The timescales involved enable the implementation of on-site or off-site prevention or protective measures. However, this does mean that any occurrence of significant fuel degradation in the spent fuel pool would directly lead to a large radioactive release. Therefore, any plant event sequence with significant degradation of the fuel assemblies stored in the spent fuel pool has to be considered for practical elimination.		
321.	Canada	48	4.12,	item e) Sentence 3	*Most plant designs in various States locate the spent fuel pool outside of the containment, given the slow kinetics of accidents likely to lead to severe damage of the fuel assemblies stored in the spent fuel pool. The timescales enable the implementation of on-site or off-site prevention or protective measures.* ~~*This option*~~ *Locating spent fuel storage outside of containment is considered as the best choice in the decision making process compared to the additional costs and operational constraints if the spent fuel pool were* ~~*also*~~ *located in the reactor building. However, this does mean that any occurrence of significant fuel degradation in the pool would directly lead to a large radioactive release. Therefore, any accident sequence with significant degradation of the fuel assemblies stored in the spent fuel pool has to be considered for ‘practical elimination’.*	Which option is referred to in sentence 3? From the position of the text, it appears to be either on-site or off-site protective measures. But it probably means SFP outside containment. Later in the sentence, use of "also" implies two SFPs, one inside, one outside.		Currently footnote 16: Several plant designs locate the spent fuel pool outside of the containment, given the slow kinetics of accidents likely to lead to severe damage of the fuel assemblies stored in the spent fuel pool. The timescales involved enable the implementation of on-site or off-site prevention or protective measures. However, this does mean that any occurrence of significant fuel degradation in the spent fuel pool would directly lead to a large radioactive release. Therefore, any plant event sequence with significant degradation of the fuel assemblies stored in the spent fuel pool has to be considered for practical elimination.		
322.	UK	47	Page 23	Footnote 5	Suggest “Therefore, any accident sequence with significant degradation of the fuel assemblies stored in the spent fuel pool is likely to be a candidate for ‘practical elimination’	Paragraph 4.5 is not as definitive as footnote 5 (it says “it may be necessary…..). Paragraph 4.16 is also less definitive than the footnote (it says “should be considered in the identification process….”, not “has to be”)				
323.	UK	48	Page 23	Footnote 5	Delete 3rd sentence starting “This opinion is….”	This is expressing an opinion on design choices for spent fuel storage which is not appropriate for this guide.	X			
324.	Egypt	6	4.12		(c) Severe accident sequences that could lead to late containment failure such as	Wording/editorial issues			X	The term is basemat penetrations.

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					(i) Basement ~~Basemat~~ penetration or containment					
325.	Egypt	7	4.12		(d) Severe accident with containment bypass such as: (i) ....As the containment function might be jeopardized ~~jeopardised~~ by the initiating e	Wording/editorial issues				Considered by the technical editor
326.	Japan	16	4.13		The classification and grouping in para. 4.12 is consistent with the recommendations provided in SSG-53 [5] and SSG-2 (Rev. 1) [8], highlighting some examples of severe accident conditions for practical elimination consideration.	Should be specified the para. number for the safety guide. Some references have been already specified.			X	Not considered necessary, but references to paragraphs in SSG-2 (Rev. 1) and SSG-53 could be presented.
327.	WNA	14	4.14		A phenomenological (top-down) approach, which considers any phenomena that might challenge the confinement safety function before or in the course of a severe accident	In the case of containment bypass (VLOCA), the confinement function can be challenged before getting into a severe accident		4.16(a) A phenomenological (top-down) approach, in which phenomena are considered that might challenge the confinement function before or in the course of a severe accident, in order to define a comprehensive list of plant event sequences, i.e. as listed in para. 4.14;		
328.	Egypt	8	4.15		…(e.g. start-up, power operation, shutdown, refueling ~~refuelling~~, maintenance)…	Wording/editorial issues				Considered by the technical editor
329.	France	59	4.17		Consider deletion	This typology could be usefull in another context but provide some confusion here	X	Text deleted		
330.	Canada	49	4.17		Delete paragraph.	This paragraph adds nothing to the more comprehensive description in para 4.12.	X	Text deleted		
331.	WNA	15	4.17		~~Three~~ Four types of scenario can be considered: (…) Type IV: accident scenarios that include confinement function degradation (or absence) and that may escalate to severe fuel degradation	§ some events listed in 4.12 (d) and (e) are of this type IV				Text deleted
332.	UK	49	4.18 & 4.19		For 4.18 and 4.19, replace with the following suggested text: "To achieve the objectives of practical eliminations, designers of new NPPs will need to consider an appropriate short list of accident scenarios, and undertake assessment aimed at identifying design and operational features that could be implemented, either for prevention or for limitation of the consequences of the severe accident condition." For para 4.27, suggest replace with: "The overall effectiveness of the provisions identified by the designer to practically eliminate large or early releases should be demonstrated through a safety assessment which includes engineering judgement and deterministic and probabilistic analyses. Some of the categories of conditions	The distinction between the objectives of the section "IDENTIFICATION AND ASSESSMENT OF SAFETY PROVISIONS FOR PRACTICAL ELIMINATION" and "DEMONSTRATION OF 'PRACTICAL ELIMINATION'" is currently not clear. Both sections talk about assessment. Para 4.19 says "In this assessment and later in the demonstration of 'practical elimination' of a severe accident condition…..", suggesting demonstration is something different from assessment. Para 4.27 says "The demonstration of practical elimination can be considered as part of the design and safety assessment process…."		Accepted for 4.18 & 4.19 4.19 Following the identification of relevant event sequences, and grouping them into a smaller set of plant conditions, as the next step, the designer should undertake an assessment aimed at identifying safety provisions in the form of design and operational features that could be implemented for demonstrating the practical elimination of each relevant plant event sequence. In this assessment, the following aspects should be considered:… Accepted for 4.27 as: 4.27 The overall effectiveness of the safety provisions identified by the designer to demonstrate practical elimination should be demonstrated through a safety assessment that includes engineering judgement, deterministic analyses and		

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					defined in para. 4.12 for the demonstration of practical elimination entail very severe challenges to the integrity of the physical barriers for radionuclide retention and necessitate specific design and operation provisions for their practical elimination. ~~The demonstration of practical elimination can be considered as part of the design and safety assessment process, including~~ It should also consider the necessary inspection and surveillance processes required during manufacturing, construction, commissioning and operation.	It is not clear who is doing the different aspects, when, with what scope, and for what purpose. It is suggested that the first section is concerned with design, whilst the second section is the safety assessment showing the adequacy of the 'final' design.		probabilistic assessments. The demonstration of practical elimination should be conducted as part of the design and safety assessment process for the plant, including the necessary inspection and surveillance processes during manufacture, construction, commissioning and operation.		
333.	France	60	4.19		The assessment aims at identifying design and operational features that could be implemented~~, either for prevention or for limitation of the consequences of the severe accident condition~~. In this assessment and later in the demonstration of 'practical elimination' of a severe accident condition, the following should be considered : … ~~I The independence of design provisions from the capability for on-site actions or use of off-site staff and equipment.~~	Limitation of consequences is not part of practical elimination Not understandable bullet	X	4.19 Several plant designs locate the spent fuel pool outside of the containment, given the slow kinetics of accidents likely to lead to severe damage of the fuel assemblies stored in the spent fuel pool. The timescales involved enable the implementation of on-site or off-site prevention or protective measures. However, this does mean that any occurrence of significant fuel degradation in the spent fuel pool would directly lead to a large radioactive release. Therefore, any plant event sequence with significant degradation of the fuel assemblies stored in the spent fuel pool has to be considered for practical elimination. (g) Avoiding the need to conduct on-site actions or use off-site personnel or equipment.		
334.	ENISS	40	4.19		Reword item (e): The independence of design provisions from the capability for on-site actions or use of off-site staff and equipment.	For clarification. It seems that some words are missing in the sentence to understand the point being made. Unable to suggest an alternative!	X	4.19 … (g) Avoiding the need to conduct on-site actions or use off-site personnel or equipment.		

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						May be worth to clarify what has to be independent from what: DEC versus DBA or/and Non-permanent equipment versus on site equipment?				
335.	Germany	37	4.19	Add point after (b)	The ability of the safety provision to provide sufficient margins for dealing with uncertainties.	This point is mentioned in SSG-2 (7.70). It should already be considered in the identification-process.		4.19 … (d) The capability of safety provisions to provide sufficient margins for dealing with uncertainties and to avoid cliff edge effects;		
336.	Italy	24	4.19	c	(c) The potential drawbacks, that might not be immediately apparent, of additional provisions introduced;	Meaning of the sentence is not clear. A possible rephrasing is proposed.	X	4.19 (e) Potential drawbacks of safety provisions, which might only become evident after the plant is put into operation (e.g. operational constraints or spurious actuations);		Considered during technical edition
337.	France	61	4.21		This identification aims at defining several options to be submitted to the decision-making process for establishing reasonably practicable design and operational provisions to achieve practical elimination. ~~This results in a design with a consistent and robust combination of lines of defence in depth~~	The concept of line of defence is not defined and reference to DiD is not adequate here	X	4.21 The designer should establish a decision making process for determining reasonably practicable safety provisions to achieve practical elimination. Several options for safety provisions should be developed and submitted to the decision making process.		
338.	Japan	17	4.21		This identification aims at defining several options to be submitted to the decision-making process for establishing reasonably practicable design and operational provisions to achieve practical elimination. ~~This results in a design with a consistent and robust combination of lines of defence in depth~~.	The last words “combination of lines of defence in depth” are not clear. The last sentence should be deleted otherwise replaced by “lines of defence” to “levels of defence in depth”. The same comment is for para. 3.64.	X			
339.	France ENISS	62 41	4.22		in-operation monitoring	Separate the 2 words -in and operation).	X	4.22 …In applying the engineering design rules and technical requirements, where relevant, appropriate testing should be applied, operational procedures should be followed, and, in operation, surveillance as well as in-service testing and inspection should be conducted.		
340.	Canada	50	4.22	Sentence 2	*It should be verified that the corresponding appropriate engineering design rules and technical requirements have been followed to ensure that they would ~~confidently~~ achieve their safety function, under the prevailing conditions, e.g. the harsh environmental conditions associated to a severe accident.*	"Confidently" used this way is usually interpreted as "with high confidence". The level of confidence appropriate for the provisions would be those that apply to the plant state in question. E.g. high confidence for DBA, best-estimate for DEC. See SSG-2.	X			
341.	UK	53	4.22		Suggest replace the 1st sentence with: “The design provisions considered in practical elimination assessments should be identified on a case-by-case, and, where relevant, associated to the appropriate level of defence in depth or plant state at which the sequence of events would be interrupted to prevent unacceptable consequences.”	A repeat of comment made by the UK at Step 7 prior to MS comment stage: “*The design of provisions for practical eliminations*…….”. This reads like some design provisions are to be practically eliminated, rather than being there to practically eliminate large or early releases.		4.23 Safety provisions for demonstrating practical elimination of some severe accident conditions could include operational provisions as well as design provisions, and as such they could involve the performance of operator actions (e.g. the opening of primary circuit depressurization valves to prevent high-pressure core melt conditions). In such cases, a human factor assessment should be part of the justification supporting any claim for high reliability of operator		

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								actions. The human factor assessment should address the following:		
342.	France ENISS	63 42	4.23		(The detrimental impact on safety of spurious opening should be taken into account in the design.)	This refer to previous text under parenthesis, to be consistent this should also be under parenthesis.	X	Text deleted		
343.	UK	50	4.23		Change to: “The detrimental effect on safety of any such spurious actions should be…..”	As worded this is restricted to spurious opening of a depressurisation valve, but should be more generally applicable.		Added in 4.19 as 4.19 … (e) Potential drawbacks of safety provisions, which might only become evident after the plant is put into operation (e.g. operational constraints or spurious actuations);		
344.	Japan	18	4.23		Design provision and operational provision for practical elimination of some severe accident conditions could require human actions to be performed (e.g. the opening of primary circuit depressurization valves to prevent high-pressure core melt conditions). In this case a human factor assessment should be part of the justification needed to support any claim for high reliability of operator actions. The detrimental impact on safety of human action errors ~~spurious opening~~ should be taken into account in the design.	Clarification. Human errors which should be taken account are not only “spurious opening (of primary circuit depressurization valves)” but all errors of human actions.		Text deleted		Spurious opening is not meant to be a human error. The subject here is that the possibility that a the valves could open spuriously when not required should be also taken into account
345.	India	20	4.23	Line 2	Local actions (including dependent actions)	In the severe accident scenario, dependent human actions play major role so this aspect needs to be highlighted.				Not understandable and clear It can only make the message complicated All human actions are dependent on something (other actions, time available, instrumentation, procedures and several performance shaping factors).
346.	Italy	25	4.24	1	[…] claimed to contribute towards the “practical elimination” […]	Typo	X			Considered during technical edition
347.	Germany	38	4.24		Some design and operational provisions claimed to contribute towards for the “practical elimination” of some severe accident sequences could be vulnerable to potential human errors prior to the accident. This type of human error could cause latent risks to be introduced that might prevent successful operation when called upon during an event or accident. In such a case, the SSCs used to deliver the action should be subject to relevant operational provisions (e.g. periodic testing, in-service inspections,	Please make clear that these are examples of measures against human errors and not vice versa		4.24 Some safety provisions claimed to contribute towards the practical elimination of some event sequences could be vulnerable to human errors that might have occurred prior to the onset of the accident. Such human errors could introduce latent risks that might prevent successful operation of a system or component when it is called upon during an event or accident. In such cases, the system or component used to perform the action should be subject to relevant operational provisions (e.g. periodic testing, in-service inspection and		

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					commissioning tests following maintenance activities, periodic system alignment checks) to limit the risk from this type of human error ~~(e.g. periodic testing, in-service inspections, commissioning tests following maintenance activities, periodic system alignment checks)~~.			surveillance, qualification tests following maintenance and periodic system alignment checks) to limit the risk from human errors of this type.		
348.	Canada	86	4.25 and 2.28		Combine these paragraphs to eliminate repetition.	Paras 4.25 and 4.28 are almost the same.	X			
349.	France ENISS	64 43	4.29		For each accident sequence group considered for 'practical elimination', an assessment has to be performed to demonstrate the acceptability of the design.	Grouping of sequences to make demonstration manageable is identified in para 4.12	X	4.29 For each group of event sequences considered for practical elimination, an assessment should be performed to demonstrate the effectiveness of the associated safety provisions.		
350.	Italy	26	4.31	1		Comment: as described in the IAEA Safety Report Series No.52 (2008) different sources of uncertainties are present. Therefore, it should be considered not only "Uncertainties due to limited knowledge of some physical phenomena, in particular those resulting from severe accident phenomena ..." but also all the other possible sources of uncertainties. This should be considered in a revised paragraph.			X	Here (para 4.31) we focus only those uncertainties related to lack of knowledge. Other uncertainties are considered in para. 4.40
351.	Canada	51	4.32		Suggest a footnote that explains that the "high level of confidence" will not be as high as that applicable to the frequencies of DBAs.	Although this comes from SSR-2/1, it should not be overemphasized. It is impossible to achieve that same "high" level of confidence as for the likelihood of more frequent accidents such as DBA. Similarly, we cannot have a DBA-level of high confidence in predictions of consequences for accidents with extreme conditions and poorly understood phenomena. We must acknowledge that a DBA level of confidence is not possible.			X	The acceptable "high level of confidence" for the demonstration of PE depends on the national regulatory authority. The IAEA should not substitute the national regulatory authority in that interpretation.
352.	Canada	87	4.33		Change to "to the extent practicable" "*4.33 The demonstration of very low likelihood with a high level of confidence should rely on the assessment of engineering aspects, deterministic considerations, supported by probabilistic considerations to the extent* ~~*possible*~~ *practicable, taking into account …*"	Use of "to the extent possible" means that everything possible must be done, even if it is not practicable. It would perhaps be possible to perform severe accident experiments at full scale. But it would likely not be practicable.	X	4.35 The demonstration that certain plant sequences are extremely unlikely to occur should rely on the assessment of engineering aspects, deterministic considerations, supported by probabilistic considerations to the extent practicable, taking into account the uncertainties due to the limited knowledge of some physical phenomena.		
353.	Canada	52	4.33		Change "reliable prediction" to "best estimate prediction".	As for 4.32, it will be impossible to achieve the level of reliability we expect in DBA analysis. This must be acknowledged. SSR-2/1 allows best-estimate analysis for DEC. For accidents more severe than DEC (requiring PE), a best-estimate	X	4.32 Computer codes and calculations used to support the demonstration of practical elimination should be verified and validated and models used should reflect best understanding of the physical phenomena involved so as to provide acceptable prediction of the event sequences and the		

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						approach can be used. SSG-2 acknowledges the difficulties: "*7.67. Analysis of severe accidents should be performed using a realistic approach (Option 4 in Table 1, Section 2) to the extent practicable. Since explicit quantification of uncertainties may be impractical due to the complexity of the phenomena and insufficient experimental data, sensitivity analyses should be performed to demonstrate the robustness of the results and the conclusions of the severe accident analyses.*" This difficulty is acknowledged in DS508 para 4.39. It would be better not to set unattainable expectations in 4.32 and 4.33 only to acknowledge later that they cannot be met.		phenomena involved. Section 5 of SSG-2 (Rev. 1) [9] provides recommendations on the use of computer codes for deterministic safety analyses.		
354.	Japan	19	4.33		Computer codes and calculations used to support the demonstration of 'practical elimination' should be verified and validated, and reflect best knowledge so as to provide reliable prediction of the accident sequences and the involved phenomena.	Completeness. Verification and validation is the crucial concept for computer codes and calculation stated in SSR-2/1 (Rev. 1) para. 5.47 (a).	X			
355.	UK	51	4.33		Add to second sentence "…(SSG-2 provides further guidance on use of computer codes for deterministic analysis).	For completeness	X	Added at the end of para 4.32		
356.	Canada	88	4.35	2nd sentence	Revise for clarity.	Needs revision for clarity. What "dedicated condition"? "To assess" what?	X	4.36 The demonstration that an event sequence can be practically eliminated should consider the following, as applicable…		
357.	France	65	4.36		In practice, the physical impossibility approach is limited to very specific cases.… .~~ An example could be the effect of heterogeneous boron dilution for which the main protection is provided by ensuring a negative reactivity coefficient for all possible combinations of the reactor power and coolant pressure and temperature. In this case, physical impossibility applies only to a prompt reactivity insertion accident.~~	The example does not seem relevant	X	4.34. …An example is the practical elimination of the effect of heterogeneous boron dilution, for which the main protection is provided first by injecting a limited volume of non-borated water which does not allow that effect to happen and second because of the negative reactivity coefficient for all possible combinations of the reactor power and coolant pressure and temperature. In this case, only a prompt reactivity insertion accident could be considered physically impossible.		
358.	France	66	4.37		~~The expression 'extremely unlikely' is by definition a probabilistic notion.~~. Although…	This too straightforward affirmation is disputable and provide non guidance		4.35 The demonstration that certain plant sequences are extremely unlikely to occur should rely on the assessment of engineering aspects, deterministic considerations, supported by probabilistic considerations to the extent practicable, taking into account the uncertainties due to the limited knowledge of some physical phenomena.		

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359.	Canada	89	4.37		Delete text in brackets. "*4.37 When the accident sequence to be 'practically eliminated' is the result of an accident sequence where the confinement function is degraded before the core melt, then core melt has to be prevented with a high degree of confidence. This means that, at least, the usual lines of defence in depth should be implemented ~~(AOO, DBA and DEC without fuel degradation)~~ and enhance them when necessary.*"	AOO, DBA and DEC are not levels of defence in depth, they are plant states. DEC without fuel degradation is not even a plant state.		4.42 If the event sequence to be practically eliminated is the result of an event sequence in which the confinement function degrades before core melt occurs, then it should be demonstrated, with a high degree of confidence, that core melt will be prevented. This means that, at least, the usual levels of defence in depth should be implemented (i.e. for anticipated operational occurrences, design basis accidents and design extension conditions without fuel degradation) with enhancements, as necessary, to prevent design extension conditions with core melt.		The text was modified to avoid ambiguity among plant states and defence in depth levels.
360.	India	21	4.37	Line 1	The expression 'extremely unlikely' is by definition a probabilistic notion. The quantitative target for these types of events could be less than 1E-7 per year.	a) Inclusion of a quantitative probabilistic target will be useful, though may not be may not be a single criterion. Based on the current experience, a value of 1E-7 per year could be considered as benchmark.				Providing a frequency target is misleading since the approach is to strive for providing first a justification of physical impossibility and later use the probabilistic insights to substantiate the assumptions considered.
361.	France	67	4.38		The demonstration of very low likelihood with a high level of confidence should rely on the assessment of engineering aspects, deterministic considerations, supported by probabilistic considerations to the extent possible, taking into account the uncertainties due to the state of knowledge of some physical phenomena. The demonstration for a condition to be 'practically eliminated' should consider the following, as applicable: (a) ~~The several lines of defence consisting~~ an adequate set of equipment and organisational provisions; (b) The robustness of these provisions ~~of these lines of defence~~ (e.g. adequate margins, adequate reliability, qualification against operation conditions); (c) The independence between these provisions ~~lines of defence~~ (i.e. adequate combination of redundancy and physical separation, diversity, functional independence	"line of defence" is an inadequate term that provide a fuzzy potential link with DiD		4.36 The demonstration that an event sequence can be practically eliminated should consider the following, as applicable: (a) An adequate set of safety provisions, including both equipment and organizational provisions; (b) The robustness of these safety provisions (e.g. adequate margins, adequate reliability, qualification for the operational conditions); (c) The independence between these safety provisions (i.e. an adequate combination of redundancy, physical separation, diversity and functional independence).		
362.	France ENISS	68 44	4.39		Deterministic analysis of severe accidents should be performed using a realistic approach (see Option 4 in Table 1, Section 2 of SSG-2 (Rev. 1) [8]) to the extent practicable. Because	Practical elimination by demonstration of extreme unlikelihood with a high level of confidence implies a conservative approach rather than a best estimate one and should be recognised as an approach. This		4.37 Deterministic analyses of severe accidents should be performed using a realistic approach (see Option 4 in table 1 of SSG-2 (Rev. 1) [9]), to the extent practicable. Because explicit		

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					explicit quantification of uncertainties may be impractical due to the complexity of the phenomena and insufficient experimental data, sensitivity analyses should be performed to demonstrate the robustness of the results and the conclusions of the severe accident analyses. Sensitivity studies could also be used to confirm the adequacy of conservative bounding analysis.	could be argued to be more appropriate particularly when sequences are being grouped as discussed in para 4.12.		quantification of uncertainties might be impractical owing to the complexity of the phenomena and insufficient experimental data, sensitivity analyses should be performed to demonstrate the robustness of the results and to support the conclusions of the analyses. Sensitivity studies could also be used to confirm the adequacy and representativeness of the selected severe accidents considered for the bounding analysis.		
363.	France	69	4.40		The decision whether or not to establish probabilistic targets to support the 'practical elimination' of accident sequences that could lead to unacceptable releases, falls under the responsibility of the regulatory body. When it is claimed that a particular accident condition has been practically eliminated ~~on the basis~~ with the support of probabilistic arguments, it needs to be taken into account that the cumulative contribution of all the different cases must not exceed the target for large or early release frequency where such a target has been established by the regulatory body	Not consistent with requirement, it could not be done only on this basis		4.38 If probabilistic arguments are used to support a claim that a particular event sequence has been practically eliminated, it should be ensured that the cumulative contribution of all the different event sequences considered does not exceed the target frequency for early radioactive releases or large radioactive releases, if such a target has been claimed by the designer or operating organization in the safety assessment of the plant or has been established by the regulatory body.		
364.	UK	52	4.40		Propose deleting the first sentence of para 4.40 and modifying the second: "~~The decision whether or not to establish probabilistic targets to support the 'practical elimination' of accident sequences that could lead to unacceptable releases, falls under the responsibility of the regulatory body.~~ When it is claimed that a particular accident condition has been practically eliminated on the basis of probabilistic arguments, it needs to be taken into account that the cumulative contribution of all the different cases must not exceed the target for large or early release frequency where such a target has been claimed by the NPP designer / operator in its safety assessment report or established by the regulatory body.	Although para 4.37 rightly says "demonstration of practical elimination cannot be approached only probabilistically", it does point out that 'extremely unlikely' is a probabilistic notion and that probabilistic targets do have a role to play. Therefore, whilst probabilistic targets are not an option, they are an important 'leg' of any safety argument (but not the totality of the arguments). Whether or not a regulatory body sets a specific probabilistic target for a country should not be the determining consideration of whether probabilistic targets is considered are part pf the designer's/operator's arguments. ONR would not set a probabilistic target. It seems unlikely many other regulatory bodies would have a specific target in such a developing area when, for example, IAEA has declined to set a target in SSR2/1 or this guide. Most modern NPP designs are intended for international deployment. It would be expected that claims of practical elimination would be put forward for a design (with a probabilistic claim), independent of the regulatory regime it is being proposed for.				

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365.	Canada	53	4.41	Sentence 1	Clarify the text.	Not clear what model is referred to. Is this an analytical model of accident progression or a probabilistic model to combine frequencies of all contributing "failures" leading to the large or early release that must be eliminated? Potentially, it could mean either or both.		4.35 The demonstration that certain plant sequences are extremely unlikely to occur should rely on the assessment of engineering aspects, deterministic considerations, supported by probabilistic considerations to the extent possible, taking into account the uncertainties due to the limited knowledge of some physical phenomena. Although probabilistic targets can be set (e.g. frequencies of core damage or of radioactive releases), the demonstration of practical elimination cannot be approached only by probabilistic means. Probabilistic insights should be used only in support of deterministic and engineering analyses. Meeting a probabilistic target alone is not a justification to exclude further deterministic and engineering analyses and possible implementation of additional reasonable safety provisions to reduce the risk. Thus, the low probability of occurrence of an accident with core damage is not a reason for not protecting the containment against the conditions generated by such an accident. In contrast, design extension conditions with core melting are required to be postulated in the design, in accordance with Requirement 20 of SSR 2/1 (Rev. 1) [1].		
366.	Germany	39	4.41	First sentence	… The validity of the model used should be checked against the dedicated condition to assess. Assumptions made for the proof must be well justified and validated.	Adding this point makes it clearer that this part of the validation needs special attention.		4.39 The validity of any probabilistic models used should be checked against the event sequence at hand. Assumptions made in support of this check should be well justified and validated.		
367.	WNA	16	4.43	(addition)	When the accident sequence to be 'practically eliminated' is the result of an accident sequence where the confinement function is degraded before the core melt, then core melt has to be prevented with a high degree of confidence. This means that, at least, the usual lines of defense in depth should be implemented (AOO, DBA and DEC without fuel degradation) and that additional prevention should also be implemented.	This family (type IV) is the most important one and demonstration should rely on existing demonstration. Only a complement is necessary.		4.42 If the event sequence to be practically eliminated is the result of an event sequence in which the confinement function degrades before core melt occurs, then it should be demonstrated, with a high degree of confidence, that core melt will be prevented. This means that, at least, the usual levels of defence in depth should be implemented (i.e. for anticipated operational occurrences, design basis accidents and design extension conditions without fuel degradation) with enhancements, as necessary, to prevent design extension conditions with core melt.		
368.	WNA	17	5		the title of section 5 should also be changed in the table of content			5. IMPLEMENTATION OF DESIGN PROVISIONS FOR ENABLING THE USE OF NON-PERMANENT EQUIPMENT FOR POWER SUPPLY AND COOLING		
369.	India	22	Heading before 5.1		IMPLEMENTATION OF DESIGN ~~PROVITIONS~~ PROVISIONS FOR ENABLING THE USE OF	Editorial	X			

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					NONPERMANENT EQUIPMENT FOR POWER SUPPLY AND COOLING					
370.	Israel	1	Table of contents and section 5 (par. 5.1)		COMMENTS Section 5 in the table of contents is named: *Minimization of the Radiological Consequences of Very Unlikely Conditions Exceeding the Plant Design Basis*, while in the text (preceding paragraph 5.1) this section is named: *Implementation of Design Provisions for enabling the Use of Non-Permanent Equipment for Power Supply and Cooling* (with misspelling of Provisions). Following the context of the text of section 5, it seems that the table of contents has to be corrected.	Reason Clarity / Editorial	X	Table of contents updated at the end of the drafting the document,		
371.	ENISS	45	5.1		As an application of SSR-2/1 requirement 14, the design basis of items important to safety at nuclear power plants ~~is~~ should be established taking into account the most limiting conditions under which they need to operate or maintain their integrity. This includes the conditions resulting from internal and external hazards. The external hazards and relevant combinations to be considered, as per requirement 17 of SSR-2/1 are identified and their relevant severity to achieve adequate protection of the public and the environment is defined as part of the site evaluation (SSR-1). A monitoring (req 28 of SSR1) over the plant lifetime is also required to identify potential evolutions (climate change) to confirm the plant design or anticipate the need for enhancements.	The sentence does not introduce the need for an appropriate initial definition of the severity of external hazard through the site evaluation, just taking it as a given. This is not so obvious and has to be reminded. Indeed, a good starting point in the definition of a "strong" design is a key point in this section of the document. It's easy and easier to improve a "weak" design. Before requiring more, it's important to clarify the starting point.	X	5.1 As an application of Requirement 14 of SSR-2/1 (Rev. 1) [1], the design basis for items important to safety should be take into account the most limiting conditions under which they need to operate or maintain their integrity. This includes the conditions resulting from external natural hazards. In accordance with Requirement 17 of SSR-2/1 (Rev. 1) [1], the effects of external hazards and relevant combinations of hazards are required to be evaluated. This is done is as part of the site evaluation for the plant (see IAEA Safety Standards Series No. SSR-1. Site Evaluation for Nuclear Installations [16]).		Considered during the technical edition
372.	Italy	27	5.1	4	[…] some conditions, exceeding the margins of the design of some SSCs, arise […]	Sentence is convoluted	X			
373.	France	70	5.1		~~The design basis of items important to safety at nuclear power plants is established taking into account the most limiting conditions under which they need to operate or maintain their integrity. However, it is possible, although very unlikely for a well designed nuclear power plant, that some conditions arise that exceed the margins of the design of some SSCs, thus impairing the fulfilment of safety functions. This is particularly important for the case of natural hazards, for which the occurrence of hazards of a magnitude that exceeds the safety~~	This article is out of scope of chapter 5 considering chapter 1. Moreover, its wording is not consistent with SSR-2/1	X			

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					~~margin of the most vulnerable SSC important to safety is generally a matter of probability. There have been cases in which some external natural hazards, such as extreme earthquakes, floods and tsunamis have exceeded the levels considered for the design as a result from the site evaluation. Paragraphs 5.21 and 5.21.A of SSR-2/1 (Rev. 1) [1] require sufficient margins against external hazards for such cases in the design7~~					
374.	UK	54	5.2		Add the following to the start of the 1st sentence: "To provide resilience against natural hazards exceeding those considered in the design,….."	Link the need to consider beyond design basis hazards (para 5.1) with the subject of this section which is the consideration of non-permanent equipment. This is also the subject of footnote 8 which should be linked to this paragraph.	X	5.3 To provide resilience against levels of external hazards exceeding those considered for design, several requirements are established in SSR-2/1 (Rev. 1) [1] regarding the inclusion of features in the design to enable the safe use of non-permanent equipment for the following purposes :		
375.	Canada	54	5.3		Provide a full quote from SSG-2.	Better to include the whole of SSG-2 para 7.51 or 7.64 as this establishes that non-permanent equipment <u>can be credited</u> in the long term. "*7.51. Non-permanent equipment should not be considered in demonstrating the adequacy of the nuclear power plant design. <u>Such equipment is typically considered to operate for long term sequences and is assumed to be available in accordance with the emergency operating procedures or accident management guidelines.</u> The time claimed for the availability of non-permanent equipment should be justified.*"		5.5 … The aim of the use of non-permanent equipment is to restore safety functions that have been lost, but it should not be the regular means for coping in the short term phase for design basis accidents or for design extension conditions (see also paras 7.51 and 7.64 of SSG-2 (Rev. 1)).		
376.	UK	55	5.3		Suggest changes to 2nd & 3rd sentences: "Consistent with the intentions of para. 7.51 of SSG-2 (Rev. 1), the aim of the use of such equipment is to restore safety functions that have been lost, but not to be the regular means to achieve these functions in accident conditions." Delete 3rd sentence ~~"Non-permanent equipment should not be credited in demonstrating the adequacy of the nuclear power plant design (see para. 7.51 of SSG-2 (Rev. 1) [8]."~~	The last sentence refers to the statement in SSG-2 that non-permanent equipment should not be considered for demonstrating the design. This sentence does not seem to be consistent with much of this section. For example, 5.10, 5.11 & 5.12 recognise the potential benefits (<u>if justified</u>) of non-permanent equipment in addition to fixed installed equipment, and that it can be credited - ONR would support this position. It also includes footnote 8, but this appears to be referring to the requirements from SSR 2/1 for connection points, not on whether the associated equipment can be credited. The issue		5.5 Non-permanent equipment is primarily intended for preventing unacceptable radioactive consequences in the long term phase of accident conditions and after very rare events (e.g. natural external hazards exceeding the levels considered for the design, derived from the hazard evaluation for the site) for which the capability and availability of design features installed on-site might be affected. The aim of the use of non-permanent equipment is to restore safety functions that have been lost, but it should not be the regular means for coping in the short term phase for design basis accidents or for design extension conditions (see also paras 7.51 and 7.64 of SSG-2 (Rev. 1)).		Accepted but there is a need to mention that such equipment should not be considered during the short term phase of the accident sequences for DEC.
377.	Israel	2	Paragraphs 5.3, -5.10, 5.11, 5.12		These paragraphs address the extent of "credit" which can be given to non-permanent equipment. We would like to suggest to reconsider some of the phrasings in these	Clarity				Appropriate wording according to the IAEA Safety Standards in use and IAEA Safety

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			and 5.16		paragraphs in order to avoid some possible misunderstanding on that issue of "credit": In paragraph 5.3 we find: *Non-permanent equipment should not be credited in demonstrating the adequacy of the nuclear power plant design.* In paragraph 5.10: .. *an adequate balance between fixed equipment and non-permanent equipment should be implemented.* Paragraph 5.11: *The storage location of non-permanent equipment at distance from the units can be of advantage in the case of some extreme natural hazards.* Paragraph 5.12: *If non-permanent equipment is credited, its installation and use should be documented, and comprehensive training, testing and drills should be periodically conducted to maintain proficiency in the use of the equipment and associated procedures.* And finally in paragraph 5.16: Where there is high confidence of the timely connection and operation of non-permanent equipment, *their use could be credited for accident management* to prevent unacceptable radiological consequences.					Glossary was considered by the technical editor.
378.	France	71	5.4		In order to approach the implementation of design features for using non-permanent equipment, levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site should be considered and their consequences evaluated as part of the defence in depth approach. This should be done to establish accident management measures to increase the response capability of the nuclear power plant ~~so as to make accidents with harmful radiological consequences very unlikely~~	Depending on the interpretation of harmful radiological consequences, nonpermanent equipement could be fully indaequate		5.6 To meet the requirements set out in para. 5.3, levels of natural hazards exceeding those considered for design, i.e. those derived from the hazard evaluation for the site, should be considered and their consequences should be evaluated as part of the defence in depth approach. Particularly for external hazards, if the design basis for the plant is well established, it is expected that the frequency of occurrence of a natural hazard of a severity significantly exceeding the levels considered for design will be very low. However, as such frequencies are generally associated with significant uncertainties, the behaviour of structures, systems and components to loading parameters resulting from levels of external hazards exceeding those considered for the design should be well understood.		

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379.	France ENISS	72 46	5.5		However, as such frequencies are generally associated with significant uncertainties, ~~it is very important to understand the behaviour of SSCs to loading parameters resulting from~~ for levels of external hazards beyond ~~above~~ the design basis, the plant should be able to cope with the situation: • To a certain extent, on the basis of the demonstration of the behaviour (margin) of a set of SSCs (that are necessary to reach a safe state), against the resulting loading of such a situation, After the main effects of the hazards, on the basis of the use of non-permanent equipment to restore the necessary safety functions.	It may be interesting to get information on margin to identify pitfalls of cliff-edge effects, but it's not clear why this is "very important". What is important is : 1. Have a good and reasonable level of magnitude for design basis external hazard (see comment on 5.1) 2. Have a certain margin to accommodate higher levels, at least to manage a severe accident. 3. Recognise that at some point the design margin will be exceeded and the possibility to use non-permanent equipment especially off-site ones have to be enabled. Vocabulary: DS498 is using the wording "beyond design basis" Suggest to use it to be consistent. Alternative is to stick to 5.21a: "levels of natural hazards exceeding those considered for design"	X	5.7 An evaluation should be conducted to demonstrate that the plant would be able to cope with a hazard of a severity exceeding the levels considered for the design as follows: • To a certain extent, on the basis of the demonstration of the margin of a set of structures, systems and components that are necessary to reach a safe state, against the resulting loading of such a situation; • After the main effects of the hazard have passed, and/or in addition to this, on the basis of the use of non-permanent equipment to restore the necessary safety functions.		
380.	France	73	5.5		Delete footnote	Use of non permanent equipement are generally not acceptable for practical elimination (how to prenvent early releases with non permanent equipement…)	X			
381.	UK	56	5.5		Delete footnote 9: ~~"The concept of practical elimination is not applied to external hazards within the safety analysis due to the difficulties in providing a safety demonstration based on design features comparable to the full set of cases addressed in Section 4, and it is necessary to ensure in other terms that the risk of early radioactive releases or large radioactive releases as a result from extreme external hazards is very low"~~	Para 5.5 seem reasonable, but footnote 9 does not make sense. It seems to be saying that PE is not applied to external hazards, but the reasons for this are not clear (other than that there are 'difficulties'). This seems to be inconsistent with para 4.25. ONR's current expectation is that early or large releases arising form extreme external hazards should be practically eliminated to the extent possible, rather than it not being applied.	X			
382.	Canada	90	5.5		Add additional text to complete the guidance of SSG-2 Rev 1. "*5.5 Non-permanent equipment should not be credited in demonstrating the adequacy of the nuclear power plant design in the short term. It may be credited for long term for long term sequences and is assumed to be available in accordance with the emergency operating procedures or accident management guidelines. (see para. 7.51 of SSG-2 (Rev. 1) [8].*"	SSG-2 Rev 1 para 7.51 is misleading. The first sentence appears to be clear that non-permanent equipment cannot be credited. However, the following sentence contradicts it and explicitly allows credit to be taken for the operation of non-permanent equipment in long-term sequences. DS508 should not mislead by quoting selectively.		5.5 Non-permanent equipment is primarily intended for preventing unacceptable radioactive consequences in the long term phase of accident conditions and after very rare events (e.g. natural external hazards exceeding the levels considered for the design, derived from the hazard evaluation for the site) for which the capability and availability of design features installed on-site might be affected. The aim of the use of non-permanent equipment is to restore safety functions that have been lost, but it should not be the regular means for coping in the short term phase for design basis accidents or for design extension conditions (see also paras 7.51 and 7.64 of SSG-2 (Rev. 1)).		

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383.	Egypt	9	5.5		…., it is very important to understand the behavior ~~behaviour~~ of SSCs…					Considered by the technical editor
384.	France ENISS	74 47	5.6		For each relevant scenario of an external hazard beyond ~~above~~ the design basis, the evaluation should identify limitations on the plant response capability and should define a strategy to cope with these limitations.	Suggest to use the wording "beyond" as per DS498 instead of "above" or to use the terminology of SSR-2/1 req 5.21A (levels of natural hazards exceeding those considered for design, …)	X	5.8 For each relevant scenario involving an external hazard of a level beyond the design basis, the evaluation should identify limitations on the response capabilities of the plant and a strategy should be defined to cope with these limitations.		
385.	France ENISS	75 48	5.6a		A robustness analysis of a relevant set of items important to safety to estimate the extent to which those items would be able to withstand levels of natural hazards exceeding those considered for ~~their~~ design ~~basis.~~	For most SSCs, their design is not directly linked to a level of hazard (internal flooding, Tornado…) so better to stick to SSR-2/1 terminology	X	5.8(a) A robustness analysis of a relevant set of items important to safety to estimate the extent to which those items would be able to withstand levels of natural hazards exceeding those considered for design;		
386.	France ENISS	76 49	5.6b		An assessment of the extent to which the nuclear power plant would be able to withstand a loss of a large number of SSCs ~~the safety functions~~ without reaching unacceptable radiological consequences for the public and the environment	If the meaning behind "safety functions" is the loss of the three fundamental safety functions (Cooling, Reactivity, Confinement), it's hard to imagine limited consequences. Better to focus on SSC here. See also comment on 5C below.	X	5.8(b) An assessment of the extent to which the nuclear power plant would be able to withstand a loss of the safety functions without there being unacceptable radiological consequences for the public and the environment;		
387.	France ENISS	77 50	5.6c		A definition of the coping strategies to limit and mitigate the consequences of the scenarios leading to a loss of some ~~key~~ safety functions	Idem. "Key" is generally seen as "main/fundamental" safety functions.	X	5.8(c) The coping strategies to limit and mitigate the consequences of scenarios that could lead to a loss of relevant safety functions;		
388.	UK	57	5.7		Suggest 2nd sentence is deleted: ~~"However, where applicable, specific facilities and equipment, should be considered at the final stage of the design of new nuclear power plants."~~	The text is contradictory. It suggests that some aspects cannot be fully considered at the plant design stage (to be considered in later phases), but then says where applicable it should be considered at the final stage of the design. To simplify, suggest the reference to the 'final stage of design' is just removed.		5.9 Some aspects of the use of non-permanent equipment and the associated safety assessment cannot be fully considered in detail at the design stage and should be considered in the commissioning and operation stages. However, specific provisions to ensure radiation protection of operating personnel for the use of non-permanent equipment should be considered at the design stage of new nuclear power plants or during the implementation of modifications, where applicable, for nuclear power plants designed to previous standards.	X	Modified as presented
389.	France ENISS	78 51	5.7		Some aspects of the use of non-permanent equipment and the associated safety assessment addressed in this Safety Guide cannot be fully considered in detail at ~~the~~ plant design stage and should be considered in ~~more detail during~~ the commissioning and operation phases. ~~However,~~ To allow the use of non-permanent equipment, this including operating personnel protection, where applicable, specific facilities and equipment, should be considered at the final stage of the design of new nuclear power plants. The evaluation should consider the possibility that multiple units at the same site could be simultaneously affected	This is a key lesson from the Fukushima Daichi events: to allow for the use of non-permanent equipment. May be worth to emphasize SSR-2/1 here.				
390.	France	79	5.8		Consider deletion	Out of scope				It provides further recommendations related to para 5.7 and 5.8

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391.	Canada	55	5.9		The coping strategies should be defined, and the associated coping provisions should be specified and designed taking into account the ~~most unfavourable possible~~ scenarios defined according to 5.4	This is unreasonable and includes everything that is not actually impossible. SSR-2/1 para 5.17 includes: "Causation and likelihood shall be considered in postulating potential hazards." which will allow screening out of the crazy scenarios. Design is based on the "hazard evaluation for the site" which is the screened-in scenarios. "*5.21A. The design of the plant shall also provide for an adequate margin to protect items ultimately necessary to prevent an early radioactive release or a large radioactive release in the event of levels of natural hazards exceeding those considered for design, derived from the hazard evaluation for the site.*"		5.12 The coping strategies should be defined, and the associated coping provisions should be specified and designed taking into account the possible scenarios, in accordance with para. 5.8.		
392.	Germany	40	5.9		The coping strategies should be defined, and the associated coping provisions should be specified and designed taking into account the most unfavourable possible scenario ~~defined~~ according to 5.4.	Para. 5.4 is listing the scenario, no definition is provided in it		5.12 The coping strategies should be defined, and the associated coping provisions should be specified and designed taking into account the possible scenarios, in accordance with para. 5.8.		
393.	Egypt	10	5.9		…designed taking into account the most unfavorable ~~unfavourable~~ possible scenario…					Considered by the technical editor
394.	Indonesia	15	5.10	9	To make the coping strategies more reliable, an adequate balance between fixed equipment and non-permanent equipment should be implemented. This balance should be defined considering the coping time, the time for installation, flexibility of equipment for different purposes, human reliability, human resources and the total number of actions by operating personnel needed for the whole strategy. The use of permanent fixed equipment should be preferred for the implementation of short-term actions. However, use of non-permanent equipment as backup to potentially failed installed equipment, including for short-term actions, may provide innovative and diverse means to further reduce risk and should be considered. NPP with a low power output, a non-permanent system can be excluded	Consider adding 'npp with a low power output, non-permanent system can be excluded. 'at the end of Para 5.10				This is not in agreement with the requirements
395.	France	80	5.11		The use of non-permanent equipment should ~~be credited provided~~ be such that …	Consistency with art 5.3 that states that it should not be credited If the scope of this article is different, it should be clearly explain	X			

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396.	France	81	5.12		~~If~~ Where relevant non-permanent equipment ~~is credited~~, ~~its~~ installation and use should be documented, and …	Consistency with art 5.3 that states that it should not be credited If the scope of this article is different, it should be clearly explain		5.15 The installation and use of non-permanent equipment should be documented, and comprehensive training, testing and drills should be periodically conducted to maintain operator proficiency in the use of the equipment and associated procedures.		
397.	India	23	5.12		If non-permanent equipment is credited, its installation and use should be documented, and comprehensive training, ~~testing~~ surveillance and drills should be periodically conducted to maintain proficiency in the use of the equipment & associated procedures and ascertain equipment healthiness.	Periodic surveillance is important aspect. Periodic testing to ascertain healthiness of equipment				Equipment healthiness has never been used in the safety standards and it seems to refer more to monitoring or surveillance than to testing.
398.	Canada	56	5.15		*5.16 Where there is ~~high~~ reasonable confidence of the timely connection and operation of non-permanent equipment, their use could be credited for accident management to prevent unacceptable radiological consequences.*	The appropriate level of confidence is "reasonable". SSR-2/1 5.27 includes "*The main technical objective of considering the design extension conditions is to provide assurance that the design of the plant is such as to prevent accident conditions that are not considered design basis accident conditions, or to mitigate their consequences, as far as is reasonably practicable.*" Footnote 13 allows use of a "best estimate approach" to analysis of DEC.				Paragraph deleted from discussions from September meeting.
399.	India	24	Page 30		*Suggestion* It will be useful to re-draft these clauses (clauses 5.11 to 5.16.) clearly differentiating these contexts.	The clause 5.3 states "Non-permanent equipment should not be credited in demonstrating the adequacy of the nuclear power plant design" The clause 5.11- 16 mentions about "credit" for non-permanent equipment. This is essentially in the context of accident management/ restoration of safety functions. The usage of same term "credit" although in different contexts, may create ambiguity in the interpretation of this guide.		Considered in paras 5.5 Non-permanent equipment is primarily intended for preventing unacceptable radioactive consequences in the long term phase of accident conditions and after very rare events (e.g. natural external hazards exceeding the levels considered for the design, derived from the hazard evaluation for the site) for which the capability and availability of design features installed on-site might be affected. 5.13 … The use of permanent fixed equipment should be preferred for the implementation of short-term actions. However, use of non-permanent equipment should be considered as backup to fixed equipment that might fail, including for short term actions, as it can provide innovative and diverse means to further reduce risk.		
400.	France	82	5.16		~~Where there is high confidence of the timely connection and operation of non-permanent equipment, their use could be credited for accident management to prevent unacceptable radiological consequences~~	Consistency with art 5.3 that states that it should not be credited If the scope of this article is different, it should be clearly explain	X	Text deleted		
401.	WNA	18	5.16		5.16 Where there is high confidence of the timely connection and operation of non-permanent equipment, their use could be credited for accident management to prevent	Consider deleting 5.16 as it may suggest that non-permanent could be used in the safety demonstration or include a second reminder about the fact that non-permanent equipment should not be considered in				Paragraph deleted from discussions from September meeting.

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					unacceptable radiological consequences (keeping in mind that « *Non-permanent equipment should not be considered in demonstrating the adequacy of the nuclear power plant design* » (7.51 and 7.64 of SSG-2 (Rev 1) [8] regarding the availability of systems to be considered in the safety demonstration). Recommendations regarding the use of non-permanent equipment in EOPs for prevention of significant fuel degradation or in SAMGs to mitigate the consequences of significant fuel rod degradations, and in particular recommendations concerning their timely connection and operation, are provided in SSG-54, Accident Management Programmes for Nuclear Power Plants [14].	demonstrating the adequacy of the nuclear power plant design. Reference to SSG-54 should be included (by the way, SSG-54 [14] does not appear in the reference list !)				
402.	Indonesia	16	All annexes		The use of numbering style in both annexed, e.g., I.x or II.ix followed by a dot at the end of the number is inconsistent with the use of numbering style throughout the chapters	Consider using a consistent numbe5ring style throughout the document, e.g., x.y with or without a dot at the end of the number	X	Considered by the technical editor		
403.	France	83	Annex I		Annex I to be removed If not deleted, at a very minimum: • Title should be replaced by "preliminary considerations in relation with practical elimination concept • each part of annex I should be complemented with consideration of existing guidances regarding the topic they deal with (storage pool, main primary components, criticality…).	Even if annex is not integral part of a standard, this annex potentially challenges the consensus on this DS. Regarding the concerns identified in the main text of the draft, it is better not to have detailed annexes that would potentially reinforce the challenge of requirements consistency. Principle of annex was agreed during NUSSC member meeting in February 20 but it was not expected to be as such. In particular, deletion of Annex 1 is highly recommended: - It seems to be a copy-paste of an annex of TECDOC 1791 which is not a consensual document. Even if annex is not part of a standard document, having the same annex is two different document would be a misleading message - It does not consider existing guidances regarding the topic they deal with (storage pool, main primary components, criticality…).				This Annex has been agreed at the DPP and later on at the NUSSC WG meeting. Its deletions should be agreed by NUSSC. TECDOC-1791 has been the initial source of this Annex since it was already an IAEA publication that has been used as a source of information in the development of SSG-2 (Rev.1) and other IAEA publications. In addition, even though TECDOC-1791 is not a consensus document, it was exceptionally discussed by the NUSSC for its approval, which led to resolve comments of several NUSSC members.

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404.	Egypt	12	Annex I		The design shall provide for sufficient flow routes between separate compartments inside the containment. The cross-sections of openings between compartments shall be of such dimensions as to ensure that the pressure differentials occurring during pressure equalization in accident conditions do not result in unacceptable damage to the pressure bearing structure or to systems that are important in mitigating the effects of accident conditions.	It is proposed to be added in Annex I. This is an important requirement which is addressed in SSR-2/1, Para. 6.27. This should be reflected and explained in DS 508				The purpose of this safety guide is to provide recommendations related to the implementation of practical elimination of plant event sequences. Specific recommendations related to containment are presented in SSG-53.
405.	Germany	41	I-2		(c) The metal component or structure needs to be tolerant ~~of~~ for defects;…	Clarification. As option – defects-tolerant			X	According to technical editor
406.	India	25	I-2		(f) An effective in service inspection, surveillance and chemistry control programme needs to be in place during the manufacturing, construction, commissioning and the operation of the equipment to detect any defect or degradation mechanisms and to ensure that the equipment properties are preserved over the lifetime of the plant.	To avoid degradation and /or assist degradation of the material for maintaining integrity of reactor coolant system	X			
407.	Canada	57	I-7		*I-7. Reactivity accidents can be very energetic and have a potential to destroy the fuel and other barriers. The prevention of such accidents ~~needs to~~ may be ensured at the first level of defence in depth by proper design of the reactor coolant system and the core, or at level 3 by provision of two diverse, independent means of shutdown. The ~~main~~ level 1 protection ~~is~~ may be provided by the core nuclear characteristics (such as the negative power coefficient of reactivity in LWR reactors) ~~an overall negative reactivity coefficient~~ under all possible combinations of reactor power, neutron absorber concentration, coolant pressure and temperature, being such as to contribute to the practical elimination of events involving fast reactivity insertion that could otherwise challenge the acceptance criteria for DEC events, ~~thus suppressing reactor power increase during any disturbances and eliminating the reactivity hazards with help of the laws of nature (demonstration of practical elimination by impossibility of the conditions)~~. Level 3 defence in depth protection may be provided by*	Protection can also be provided at level 3, for example in PHWRs with two diverse, independent means of shutdown, together with small excess reactivity and long prompt neutron lifetimes. This can be more reliable than certain level 1 DiD provisions such as administrative controls on boron concentration or pump starting in PWR. Also, consider the potential consequences of normally screened-out sequences such as a main steam line break in a PWR if there were no shutdown before the cold water reached the core.	X	I-7. Fast reactivity accidents can be very energetic and have a potential to destroy the fuel, fuel cladding and other barriers. As far as possible, the prevention of such accidents is to be ensured at the first level of defence in depth by proper design of the reactor coolant system and the core, or at the third level of defence in depth by provision of two diverse, independent means of shutdown. I-8. The first level of defence in depth may be provided by the core nuclear characteristics (such as the negative reactivity coefficient in light water reactors), which, under all possible combinations of reactor power, neutron absorber concentration, coolant pressure and temperature, suppresses any increase in reactor power during any disturbances and eliminate any uncontrolled reactivity excursion. Therefore, this is a case of demonstration of practical elimination by physical impossibility of the event sequence.		

TITLE: DS508 - Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants

No	MS/ Org.	Comment No.	Para	Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					two independent, diverse means of shutdown of the reactor core.					
408.	Germany	42	I-7		Reactivity accidents can be very energetic and have a potential to destroy the fuel, fuel cladding and other barriers.	Clarification	X	I-7. Fast reactivity accidents can be very energetic and have a potential to destroy the fuel, fuel cladding and other barriers….		
409.	India	26	I-7		I-7. Reactivity accidents can be very energetic and have a potential to destroy the fuel and other barriers. As far as possible, the prevention of such accidents needs to be ensured at the first level of defense in depth by proper design of the reactor coolant system and the core. In most NPP designs, main protection is provided by an overall negative reactivity coefficient under all possible combinations of reactor power, neutron absorber concentration, coolant pressure and temperature, thus suppressing reactor power increase during any disturbances and eliminating the reactivity hazards with help of the laws of nature (demonstration of practical elimination by impossibility of the conditions). In some designs due to the inherent characteristics of the core, where it may not be possible to achieve the overall negative reactivity in the first level of defense, the subsequent levels of Defense in Depth should be highly reliable making the scenario highly improbable/practically eliminated.	For pressure tube based reactor type, it may not be possible achieve the overall negative reactivity with the core and RCS characteristics in the entire Power operating range. So for such designs, the systems provided in the subsequent levels of DiD should be highly reliable to **practically eliminate** the fast reactivity addition events/ scenario which have a potential to destroy multiple barriers.		I-7. Fast reactivity accidents can be very energetic and have a potential to destroy the fuel, fuel cladding and other barriers. As far as possible, the prevention of such accidents is to be ensured at the first level of defence in depth by proper design of the reactor coolant system and the core, or at the third level of defence in depth by provision of two diverse, independent means of shutdown. I-8. The first level of defence in depth may be provided by the core nuclear characteristics (such as the negative reactivity coefficient in light water reactors), which, under all possible combinations of reactor power, neutron absorber concentration, coolant pressure and temperature, suppresses any increase in reactor power during any disturbances and eliminate any uncontrolled reactivity excursion. Therefore, this is a case of demonstration of practical elimination by physical impossibility of the event sequence.		
410.	Germany	43	I-7	Line 3	… The main protection is provided by an overall negative reactivity coefficient under all possible combinations of reactor power, neutron absorber concentration, coolant pressure and temperature, thus suppressing reactor power increase during any disturbances and eliminating the uncontrolled reactivity ~~hazards~~ excursion with help of the laws of nature (demonstration of practical elimination by impossibility of the conditions).	Clarification		I-8. The first level of defence in depth may be provided by the core nuclear characteristics (such as the negative reactivity coefficient in light water reactors), which, under all possible combinations of reactor power, neutron absorber concentration, coolant pressure and temperature, suppresses any increase in reactor power during any disturbances and eliminate any uncontrolled reactivity excursion. Therefore, this is a case of demonstration of practical elimination by physical impossibility of the event sequence.		
411.	Canada	58	I-9		*The demonstration of practical elimination relies primarily on impossibility of reactivity excursions through a core design with overall small or negative reactivity coefficients…*	In PHWR, small (often near-zero) reactivity coefficients are important in these demonstrations.	X	I-10. Therefore, the demonstration of practical elimination relies primarily on impossibility of reactivity excursions through a core design with overall small or negative reactivity coefficients, supported by other design measures to avoid or limit excursions of reactivity, which can be evaluated deterministically and probabilistically as appropriate to demonstrate that the conditions are extremely unlikely to occur.		

TITLE: DS508 - Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants

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412.	Canada	59	I-10		*I-10. More complex situations could arise however if criticality can be reached during severe accidents. This has been a topic of concern in specific core meltdown scenarios in reactors using enriched fuel where the control rod material has a lower melting point and eutectic formation temperature than the fuel rods.*	This only applies to reactors using enriched fuel.	X	I-11. A more complex situation could arise however if criticality can be reached during a severe accident. This has been a topic of concern for specific core meltdown scenarios in reactors, for which the control rod material has a lower melting point and eutectic formation temperature than the fuel rods.		
413.	UK	58	I-10`		Before the last sentence add: “This could result in re-criticality of the fuel, likely resulting in a generation of additional heat on a continuing or intermediate basis, depending on the presence of water.”	To provide further information on the likely progression of a re-criticality.	X	I-11 … This could result in re-criticality of the fuel, likely resulting in the generation of additional heat on a continuing or intermediate basis, depending on the presence of water….		
414.	Canada	60	I-11		*In ~~light water~~ pressure vessel reactors, core meltdown at high pressure could cause a violent discharge of molten core material into the containment atmosphere and this would result in direct containment heating by chemical reaction…*	Direct containment heating is not a concern in PHWRs because of their pressure tube design, not because they use heavy water	X	I-12. In a pressure vessel reactor, core meltdown at high pressure could cause a violent discharge of molten corium material into the containment atmosphere…		
415.	UK	59	I-11		Change 3rd line to read: “….and this would result in direct containment heating from the hot melt and exothermic chemical reactions”.	For completeness	X	I-12 … and this would result in direct containment heating from the hot melt and exothermic chemical reactions….		
416.	ENISS	52	I-16		The conditions of the triggering of the steam explosion and the energy of explosion in various situations have been widely studied in reactor safety research programs. Although spontaneously ~~non~~triggered steam explosion seems to be ~~very~~ unlikely, the risks of steam explosion cannot be fully eliminated in all core meltdown scenarios where molten core may be dropped to water.	Non-triggered makes no sense. I presume you are referring to spontaneously triggered steam explosions i.e. without an external trigger. “very” is perhaps a bit too strong?	X	I-18. The conditions of the triggering of a steam explosion and the energy of explosion in various situations have been widely studied in reactor safety research programmes. The risks of steam explosion cannot be fully eliminated for all core meltdown scenarios in which molten core might drop to water.		
417.	UK	60	I-16		Change 2nd sentence to read: “~~Although nontriggered steam explosion seems to be very unlikely,~~ The risks of steam explosion cannot be fully eliminated in all core meltdown scenarios where molten core may be dropped to water.”	The first part of this sentence is an opinion which could be contentious. The key message is that it cannot be fully eliminated.				
418.	Canada	61	I-17		*I-17. For eliminating steam explosions that could damage the containment barrier, the preferred method is to avoid the dropping of molten core into water in any conceivable accident scenarios. …*	Does this apply to <u>in-vessel</u> relocation of molten core as well as to ex-vessel? The core catcher solution (given in I-17) only applies to ex-vessel. Is the in-vessel case treated some other way?				The intent is to provide a general description of the approach for in-vessel and ex-vessel corium cooling, which is presented in paragraph I-19, without entering into detail.

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										A sentence could be added to mention the accident management strategies related to in-vessel corium cooling.
419.	Germany	44	I-17		For eliminating steam explosions that could damage the containment barrier, the preferred method is to avoid the dropping of molten core into water in any conceivable accident scenarios. Such approach is used in some pressurized water reactors, ~~such as existing small reactors~~ where reliability of external cooling of the molten core has been proven and in some new reactors with a separate core catcher…	We suggest deleting "small reactors", as this definition is being discussed currently and is not fixed jet, the same for classification according to reactor power	X	I-19 … Such approach is used in some pressurized water reactors where reliability of external cooling of the molten core has been proven and in some new reactors with a separate core catcher….		
420.	Germany	45	I-17	Line 10	… The role of ~~PSA~~ probabilistic safety assessment in this demonstration, if there is one at all, is very limited.	This is the first and the only time where this abbreviation is used				Original text of that sentence was deleted to avoid misunderstanding
421.	UK	61	I-17		Delete the last sentence: ~~"The role of PSA in this demonstration, if there is one at all, is very limited."~~	It is understood that some practical elimination claims are heavily reliant on PSA, particularly where code validation is limited. The sentence may therefore not represent reality.	X			Original text of that sentence was deleted to avoid misunderstanding
422.	Canada	62	I-18 to I-22		Consider changing "hydrogen" to "combustible gas" where appropriate.	Hydrogen is not the only combustible gas but is the only one mentioned in paragraphs I-18 to I-22. Carbon monoxide is not mentioned until I-23.				No need since they are described in separate paragraphs.
423.	India	27	I-18		Dedicated means to prevent generation, accumulation of Hydrogen in higher concentrations, and eliminate hydrogen detonation are needed at all nuclear power plants, although different means are preferred for different plant designs.	The prevention and mitigation both aspects are important.	X	I-20 … Dedicated means to prevent the generation of hydrogen and its accumulation at critical concentrations, and to eliminate hydrogen detonation, are needed at all nuclear power plants, although different means are preferred for different plant designs.		
424.	UK	62	I-21		Change 1st sentence to read: "Consequences will be sensitive to the highest conceivable rate and the total amount of hydrogen generation inside the containment."	The inference that further research is required is not helpful to the designer.		I-23. The consequences of hydrogen combustion will depend on the highest conceivable rate and the total amount of hydrogen generation inside the containment. Some core catchers that are currently installed in nuclear power plants can significantly reduce or even eliminate ex-vessel hydrogen generation in an accident when the molten core has dropped to the catcher, and this could also considerably reduce the total amount of hydrogen generated inside the containment.		
425.	India	28	I-22		This assessment also includes the consideration of hydrogen & steam transport ~~propagation~~ and mixing inside the containment due to turbulence, density and diffusion	The transport of Hydrogen inside the containment is due to turbulence, density and diffusion, which decides the local concentrations. Due consideration should also be given to model steam transport and its addition/removal from				Agree, but this is too much detail. The text aims only provide an overview. Further

No	MS/ Org.	Com ment No.	Para	Line No.	Proposed new text	Reason	Accept ed	Accepted, but modified as follows	Rejec ted	Reason for modification/rejection
						containment atmosphere as can affect combustion characteristics significantly.				details are presented in SSG-53.
426.	ENISS	53	I-23		The design provisions for preventing hydrogen detonation need to be assessed in order to demonstrate the practical elimination of this phenomenon. This assessment also includes the consideration of hydrogen propagation and mixing inside the containment. ~~This is of particular importance in case of molten core concrete interaction when the amount of hydrogen exceeds the capacity of recombination due to lack of oxygen in the containment.~~	Delete last sentence as, when there is not enough oxygen to support the operation of the PAR (~1vol%) then there will not be enough oxygen to support a detonation (~10vol%).	X			Actually I-22
427.	UK	63	I-25		Change to: "There are several examples, from both existing plants and from new plant designs, of robust dedicated containment cooling systems that are independent of other safety systems and may be capable of supporting a demonstration to practical elimination of containment rupture by overpressure."	It may be that systems are reliant on the same power source, so should not always be considered for practical elimination.	X	I-27. There are several examples, from both existing plants and from new plant designs, of robust dedicated containment cooling systems that are independent of safety systems and might be capable of supporting the demonstration of practical elimination of containment rupture by overpressure.		
428.	Egypt	11	I-26		…The existing venting systems prevent over pressurization…					Considered by technical editor
429.	UK	64	I-26		Should be "loss" in last sentence	Typo	X			
430.	USA	4	I-26		An alternative to cooling is to eliminate the containment overpressure by venting. This is necessary especially in some boiling water reactors where the size of the containment is small and pressure limitation may be needed both in the DBA as well as in DEC with core melt. The existing venting systems prevent overpressurization at the cost of some radioactive release involved in the venting, also in the case that the venting is filtered, ~~which would be the only acceptable type for severe accidents.~~ may be acceptable strategies for severe accident management if technically justified given the risk levels and appropriate assessment of the decontamination factors for the strategy.	Severe accident water addition and management with suppression pool scrubbing were found to be acceptable for severe accident management in BWRs in the US.		I-28. An alternative to cooling of the containment is elimination of containment overpressure by means of venting. This is necessary especially in some boiling water reactors, where the size of the containment is small and pressure limitation might be needed for design basis accidents and design extension conditions with core melt. The venting systems in existing plants prevent overpressurization at the cost of some radioactive release involved in the venting, also in the case that the venting is filtered. However these might be acceptable strategies for severe accident management if technically justified given the risk levels and an appropriate assessment of the decontamination factors for the strategy.		
431.	India	29	I-27		An alternative to cooling is to eliminate the containment overpressure by filtered venting	Filtered venting reduces the releases and consequences by at least two orders.		I-28 … The venting systems in existing plants prevent overpressurization at the cost of some radioactive release involved in the venting, also in the case that the venting is filtered. However these might be acceptable strategies for severe accident management if technically justified given		The design alternative is venting the containment to protect containment integrity. Agree that filtered venting

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								the risk levels and an appropriate assessment of the decontamination factors for the strategy.		reduces the consequences.
432.	ENISS	54	I-27		Containment venting avoids a threat to ~~some peaks of pressure threatening~~ ~~the~~ containment integrity due to overpressurization, but the stabilization of the core and the cooling of the containment are still necessary in the longer term.	Venting systems are generally not effective in reducing sudden peaks because the flow area is low.	X	I-29. Containment venting avoids a risk to the containment integrity due to overpressurization, but stabilization of the core and the cooling of the containment are still necessary in the longer term.		
433.	India	30	I-28		The safety demonstration needs to be based on the capacity (w.r.t. In-Vessel Retention, Ex-Vessel melt), capability and reliability of the specific measures implemented in the design to cope with the severe accident phenomena.	The capacity which can cater to IVR as well as Ex-Vessel melt severe accidents, can only eliminate the cliff edge effect and ensure safety.				Not clear. It is not only in relation to these specific measures for molten core stabilization.
434.	Italy	28	I-33	1	[…] such as through circuits […]	Syntax	X			Considered during technical edition
435.	Japan	20	I-34		It has to be taken into account that failures of lines exiting the containment and connected to the primary system, including steam generator tube ruptures, are at the same time accident initiators, whereas other open penetrations only constitute a release path in accident conditions.	A missing word.	X			
436.	Germany	46	I-36	Last line	"… bypass or interface systems loss of coolant accidents."	Editorial	X			
437.	Italy	29	I-36	6	[…] interfacing loss of cooling […]	Typo	X			
438.	Canada	63	I-38	Item (c)	*(c) Providing sufficiently ~~redundant and~~ reliable means for pool cooling that eliminate the possibility of long lasting loss of cooling function, i.e. for the time needed to boil off the water*	It may not be absolutely necessary to provide redundant means of pool cooling, depending on the relative magnitude of the peak heat load to the water volume (i.e., if there is a long enough boil-off time, then having a makeup provision in addition to a reliable but (not redundant) cooling system may be sufficient).	X	I-39… (c) Providing sufficiently reliable means (e.g. such as applying redundancy, diversity and independence see para. 3.7 of IAEA Safety Standards Series No. SSG-63…		
439.	Germany	47	I-40		In designs where the spent fuel pool is outside the containment, the uncovering of the fuel would lead to fuel damage and a large release could not be prevented. Means to evacuate the hydrogen would prevent explosions that could cause further damages and prevent a later reflooding and cooling of the fuel. Therefore, it is necessary to ensure by design provisions that the uncovering of spent fuel elements has been 'practically eliminated'.	Clarification	X	As I-41 In designs where…		
440.	Germany	48	I-41		In some designs, the spent fuel pool is located inside the containment. In this case, even though the spent fuel damage would not lead directly to a large release, the amount of hydrogen generated by a large number of fuel elements, the easy penetration of the pool liner	Wording	X	As I-42 In some designs, the spent …		

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					by the molten fuel without means to stabilize it, among other harsh effects would eventually lead to a large release. Therefore, it is also necessary to ensure by design provisions that also in this case ~~that~~ the uncovering of spent fuel elements has been 'practically eliminated'.					
441.	France	84	Annex II		Consider deletion	Regarding the concerns identified in the main text of the draft, it is better not to have detailed annexes that would potentially reinforce the challenge of requirements consistency (even if annex is not part of the document and even if principle of annex was agreed during NUSSC member meeting in February 20). If not deleted, annex II should be modified as followed at a very minimum Following comments are alternate proposal regarding deletion of annex II				Annex II was not part of the DPP. It was proposed and agreed at the NUSSC WG of February 2020. If NUSSC agrees, it would be deleted.
442.	France	85	Annex II - title		APPLICATION OF THE GUIDANCE TO NUCLEAR POWER PLANTS DESIGNED ACCORDING TO EARLIER STANDARDS COMPARED TO SSR-2/1 (Rev. 1)	Tentative to have a title consistent with the text of the annex (II-1)		Annexes don't provide guidance. ANNEX II. APPLICATION OF THE CONCEPTS OF DESIGN EXTENSION CONDITIONS AND PRACTICAL ELIMINATIONTO NUCLEAR POWER PLANTS DESIGNED TO EARLIER STANDARDS		
443.	Canada	91	II-1,	editorial	Remove "(ENISS)" from text. Two occurrences.	II-1, editorial	X			
444.	ENISS	55	ANNEX II II.1		This implies that the capability of existing plants to accommodate accident conditions not considered in their current design basis and the practical elimination of plant conditions that can lead to early radioactive releases or to large radioactive releases need to be assessed as part of the periodic safety review processes with the objective of further improving the level of safety, where reasonably practicable.	Should be included in the PSR process where reasonable practicability is considered.	X	II-1 … This implies that the capability of existing plants to accommodate accident conditions not considered in their current design basis and the practical elimination of event sequences that could lead to an early radioactive release or a large radioactive release need to be assessed as part of the periodic safety review processes with the objective of further improving the level of safety, where reasonably practicable.		
445.	France	86	II.1		II-1. Paragraph 1.3 … ~~This implies that the capability of existing plants to accommodate accident conditions not considered in their current design basis and the practical elimination of plant conditions that can lead to early radioactive releases or to large radioactive releases need to be assessed with the objective of further improving the level of safety.~~.	Quotation of SSR-2/1 is sufficient. Rephrasing it is tricky: - what does "capability to accommodate" means regarding safety? - "improving the level of safety" is not clear and is not achieved just by assessment.	X	II-1. Paragraph 1.3 of IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [II–1] states: "It might not be practicable to apply all the requirements of this Safety Requirements publication to nuclear power plants that are already in operation or under construction. In addition, it might not be feasible to modify designs that have already been approved by regulatory bodies. For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further enhanced by		

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								means of reasonably practicable safety improvements."		
446.	ENISS	56	II.2		The concepts of design extension conditions and practical elimination of event sequences that could lead to early radioactive releases or large radioactive releases are not totally new. In fact, the last concept was ~~already~~ introduced in 1996 in INSAG 10 in the context of the development of defence in depth for new plants but using the synonym "essentially elimination". ~~the former~~ The expression "practical elimination" was subsequently used in the Safety Guide for the design of the reactor containment10, and both concepts may have been applied partially in the design of some existing nuclear power plants, although not in a systematic way. Over time, design features to cope with conditions such as station blackout or anticipated transients without scram have been introduced in many nuclear power plants. Some plant conditions to be practically eliminated have been addressed also in many designs already, although a specific demonstration in accordance with the concept of practical elimination has not been carried out.	For accuracy.				Normally we don't quote non safety standard publications. It is not adding value to introduce a new term, "essentially eliminated" when such a term was not defined. When loXing at the expression there and the explanations in SSR 2/1 and in DS508, it will be disputable that it is a synonym. It only adds confusion to a difficult topic.
447.	Germany	49	II-2	Footnote 10	INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Reactor Containment Systems for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.10, IAEA, Vienna (2004); suspended in 2020 by SSG-56.	Suggestion to complete, in order to show both – the history and the today	X	Footnote 20: See para. 6.5 of INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Reactor Containment Systems for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.10, IAEA, Vienna (2004), which has been superseded by INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Reactor Containment and Associated Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-53, IAEA, Vienna (2019) [II-2].		
448.	Germany	50	II-2		Some ~~plant conditions~~ event sequences that could lead to early radioactive releases or large radioactive releases to be practically eliminated have been addressed also in many designs already, although a specific demonstration in accordance with the concept of practical elimination has not been carried out.	These are event sequences, leading to early radioactive releases or large radioactive releases, which are to be practically eliminated		II-2 …Some event sequences that could lead to an early radioactive release or a large radioactive release have been addressed also in many designs already, although a specific demonstration of the practical elimination of such event sequences has not been carried out.		
449.	ENISS	57	II.3		~~In relation to practical elimination, a number of measures may have been taken for instance for the prevention of a break in the reactor pressure vessel, fast reactivity insertion accidents or severe fuel degradation in the irradiated fuel storage. However, a demonstration~~	Repeats points made in II.2 and implies that PSR should backfit a particular process (which is poorly defined in this guide) rather than to review the adequacy of the overall safety of the plant. Applying new processes may improve confidence	X	Text deleted		

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					~~that the existing safety provisions are sufficient to claim the practical elimination of such conditions might not have been conducted, in the way required by IAEA Safety Standards~~ ~~Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [II–1] and as recommended in this Safety Guide.~~	and identify reasonably practicable improvements but these are aids to judgement.				
450.	UK	65	II-3		Change to: "It is important to note however, that an accident condition commonly considered as a design extension condition in new nuclear power plants (e.g. station blackout or anticipated transients without scram), is only such if safety features have been introduced in the original design of the plant to mitigate its consequences. Otherwise, it would remain a beyond design basis accident. For the case of station blackout, an alternate power source capable of supplying power in due time to essential loads over a sufficient time period until external or emergency power is recovered would be such an original design safety feature. Likewise, for anticipated transients without scram, features capable of rendering the reactor subcritical in case of failure in the insertion of control rods, would need to have been included in the original design."	It seems to imply that once something is introduced to the design it becomes part of the design basis, but if not part of the design then it is outside of design basis. The wording needs sharpening up to make it clear that what is being referred to is the original design of a plant (otherwise this statement can be mis-read). It also needs to be made clearer that the final sentence is just a similar example of what is intended for the ATWT case.		II-4. However, an accident condition commonly considered as a design extension condition in a new nuclear power plant (e.g. station blackout or anticipated transient without scram), can only be considered a design extension condition for an existing nuclear power plant if safety features have been introduced in the original design of the existing plant to mitigate the consequences of this condition. For the case of station blackout, an alternate power source capable of supplying power in due time to essential loads over a sufficient time period until external or emergency power is recovered would be such an original design safety feature. Likewise, for anticipated transient without scram, additional design features capable of rendering the reactor subcritical in case of failure in the insertion of control rods would need to have been included in the original design. Without such additional design features in the original design, these accident conditions would need to be considered to be beyond the design basis of the plant.		
451.	Canada	92	II-4 **Major comment**		Delete opening 2 sentences: II-4 ~~It is important to note however, that an accident condition commonly considered as a design extension condition in new nuclear power plants (e.g. station blackout or anticipated transients without scram), is only such if safety features have been introduced in the design to mitigate its consequences. Otherwise, it would remain a beyond design basis accident~~. for the case of station blackout …	Please read SSR-2/1 Requirement 20. Nowhere does SSR-2/1 say that accident sequences can only be classified as DEC if there are mitigating systems provided for them. The DS508 text puts things backwards. SSR-2/1 says that credible events less frequent than DBAs are postulated and means of preventing the sequences or mitigating their consequences are considered in the design. If the wording used here was true, the easiest way to remove ATWS or SBO from DEC would be to remove the backup scram initiation and the alternate power supplies. No mitigating system, no DEC!	X	II-4. However, an accident condition commonly considered as a design extension condition in a new nuclear power plant (e.g. station blackout or anticipated transient without scram), can only be considered a design extension condition for an existing nuclear power plant if safety features have been introduced in the original design of the existing plant to mitigate the consequences of this condition. For the case of station blackout, an alternate power source capable of supplying power in due time to essential loads over a sufficient time period until external or emergency power is recovered would be such an original design safety feature. Likewise, for anticipated transient without scram, additional design features capable of rendering the reactor subcritical in case of failure in the insertion of control rods would need to have been included in the original design. Without such additional design features in the original design,		

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								these accident conditions would need to be considered to be beyond the design basis of the plant.		
452.	Italy	30	II-5	1	[…] it is expected […]	Missing word	X			
453.	UK	66	II-5		Should read: "There can, however, be important constraints to installation of the same type of design features as commonly implemented in the design….."	Typos	X			
454.	Germany	51	II-5		Generally, it is expected that during a periodic safety review ...	Typo.	X			
455.	France	88	II-7		~~II-7. Safety systems of existing plants were designed for design basis accidents, without account being taken of the possibility of more severe accidents. However, the conservative deterministic approaches originally followed in the design might have resulted in the capability to withstand some situations more severe than those originally included in the design basis for existing plants. As indicated in para. 3.20, for design extension conditions without significant fuel degradation, it can be acceptable for postulated initiating events less frequent than those considered for DBAs to demonstrate that some safety systems would be capable of and qualified for mitigating the consequences of such events if best estimate analyses and less conservative assumptions are used. This is a possibility for existing nuclear power plants to demonstrate the capability for mitigation as a design extension condition of events not originally postulated in the design, such as the multiple rupture of steam generator tubes.~~	This article does not comply with the title of the annex. It is a statement that existing plants with existing design may withstand some accidents not considered in its design if these accidents are studied with different rules				The paragraph complies with the title of the annex since it reflects how the margins considered in the design of safety systems in existing NPP could be assessed with less conservative rules than for DBA to demonstrate the capability of those safety systems to cope with accident conditions that were not considered in the original design.
456.	UK	67	II-7		Should read: "This is a possibility for existing nuclear power plants to demonstrate the capability for mitigation of design extension conditions, not originally postulated in the design, such as multiple rupture of steam generator tubes."	Improve wording		II-7 … For existing nuclear power plants, this is a possibility to demonstrate the capability for mitigation of design extension conditions not originally postulated in the design, such as a multiple rupture of steam generator tubes.		
457.	France	89	II-8		The consideration of external events of a magnitude exceeding the original design basis derived from the hazard evaluation for the site, as it is addressed in Section 5, is ~~a part of the safety reassessment of existing nuclear power plants that needs~~ to be considered. While for new nuclear power plants the mitigation of design extension conditions is expected to be	To be consistent with the SSR-2/1 and section 5 (non permanent equipment is not considered in safety demonstration) This statement/recommendation is not justified, not editorially consistent with SSR-2/1 and not relevant for this annex which is not related to new NPP		II-8. The consideration of external events of a magnitude exceeding the original design basis derived from the hazard evaluation for the site, as addressed in Section 5, is to be considered. While for new nuclear power plants the mitigation of design extension conditions is generally expected to be accomplished by permanent design features, and the use of non-permanent equipment is intended		

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					accomplished by permanent design features, and the use of non-permanent equipment is intended for very unlikely external events of a magnitude exceeding the original design basis derived from the hazard evaluation for the site, for existing nuclear power plants the use of non-permanent equipment with adequate connection features can be the only reasonable improvement in some cases. Relying on non-permanent equipment may be adequate provided there is a justification to demonstrate that the coping time to prevent the loss of the safety function that the equipment is intended to fulfil is long enough to connect and put into service the equipment under the conditions associated with the accident. The recommendations in this regard provided in Section 5 would be relevant. Non-permanent equipment that would be necessary to minimize the consequences of events ~~that cannot be mitigated~~ further than the minimization provided by by the installed plant capabilities needs to be stored and protected to ensure its timely availability when necessary, with account taken of possible restricted access due to external events (e.g. flooding, damaged roads) and its operability verified			only for very unlikely external events of a magnitude exceeding the original design basis, for existing nuclear power plants the use of non-permanent equipment with adequate connection features can be the only reasonable improvement in some cases. Relying on non-permanent equipment might be adequate provided there is a justification to demonstrate that the coping time to prevent the loss of the safety function that the equipment is intended to fulfil is long enough to connect and put into service the equipment under the conditions associated with the accident. The recommendations in this regard provided in Section 5 are relevant. Non-permanent equipment that would be necessary to reduce further the consequences of events that cannot be mitigated by the installed plant capabilities needs to be stored and protected to ensure its timely availability when necessary, with account taken of possible restricted access due to external events (e.g. flooding, damaged roads) and its operability needs to be verified.		
458.	Canada	64	Annex II II-8		*The consideration of external events of a magnitude exceeding the original design basis, as it is addressed in Section 5, is apart of the safety reassessment of existing nuclear power plants that needs to be considered. While for new nuclear power plants the mitigation of design extension conditions is generally expected to be accomplished by permanent design features, and the use of non-permanent equipment is intended for very unlikely external events of a magnitude exceeding the original design basis, for existing nuclear power plants the use of non-permanent equipment with adequate connection features can be the only reasonable improvement in some cases. …*	The point here is that the use of non-permanent equipment is not forbidden in new NPPs	X			
459.	Canada	93	Definition Practical elimination		The concept of "practical elimination" is applied ~~in relation to~~ to exclude from consideration in the design, plant conditions that can lead to early radioactive releases or large radioactive releases~~, for which reasonably~~	SSR-2/1 requires that "***the design shall be such*** *that the possibility of conditions arising that could lead to an early radioactive release, or large radioactive release is 'practically eliminated'*". Thus, PE is a way of excluding certain plant		Definition Practical elimination Ensuring by design that plant event sequences that could lead to an early radioactive release or a large radioactive release are either physically impossible		

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					~~practicable technical means for its mitigation cannot be implemented~~.	conditions from **further consideration in design**. This is addressed in the first proposed change. SSR-2/1 does not say or imply that PE applies only to exclude accidents “for which reasonably practicable means for its mitigation cannot be implemented”. A designer could choose to PE an accident scenario even if reasonably practicable technical means for mitigation are available. This is addressed in the second proposed change.		or are considered, with a high level of confidence, to be extremely unlikely to arise. The concept of practical elimination is applied in relation to event sequences for which reasonably practicable technical means for their mitigation cannot be implemented. Practical elimination is part of a general approach to design safety and is an enhancement of the application of the concept of defence in depth.		
460.										