Section 1
DS491 Step 7: Deterministic Safety Analysis for NPPs

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date:				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Canada 1	General	Use common abbreviations for AOO, DBA, DEC, etc.	Many commonly abbreviated terms are spelled out, making the document more wordy than necessary.		*(Editorial)* *The use of abbreviations is defined by the IAEA editors and apply to the other Safety Guides too.*		
Observer ENISS-1	General Comment	**This SG should be devoted only to methods and tools used in the deterministic safety analysis:** the scope of the document is very large (for high level safety principles, it even overlaps with SSR-2/1) and lead subsequently to a level of detail which is not homogeneous between sections. It addresses safety principles, PIEs identification and categorizing, safety criteria and acceptance criteria, analysis methods, calculation tools, ...					*The Safety Guide provides recommendations on how to meet applicable Safety Requirements.*
Canada 1	General	Use common abbreviations for AOO, DBA, DEC, etc.	Many commonly abbreviated terms are spelled out, making the document more wordy than necessary.		*(Editorial)* *The use of abbreviations is defined by the IAEA editors and apply to the other Safety Guides too.*		
Canada 2	1.3 2nd	The modifications incorporated in this Guide	Delete the marked text: it is not necessary.			X	*§1.3 refers to the changes*

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	sentence	reflect recent experience with deterministic safety analysis included in Safety Analysis Reports for ~~present reactor designs and with various applications of deterministic safety analysis of existing~~ nuclear power plants.	Without a clear definition of "present reactor designs" or "existing NPPs" it is not clear how differing requirements for the two classes will be applied. See comment on para 1.6 where this terminology leads to problems.				*incorporated in this draft compared to the published version. Applicability is indicated in SCOPE (see 1.6)*
Germany 1	1.4	1.4. The objective of this Safety Guide is to provide recommendations and guidance on performing deterministic safety analysis for designers, operators, regulators and technical support organizations. It also provides recommendations on the use of deterministic safety analysis in: (a) Demonstrating or assessing compliance with regulatory requirements; (b) Determination of the effectiveness of EOPs and SAMG measures (c) Identifying possible enhancements of safety and reliability;	A relevant application of deterministic safety analyses – especially after the Fukushima accidents – is also the determination of the effectiveness of both emergency operating procedures and preventive and mitigative severe accident management measures. Thus, the list should be expanded.		*Second sentence:* "It also indicate ~~provides recommendations on~~ the use of deterministic safety analysis in purposes such as: (a) Demonstrating or assessing compliance with regulatory requirements; (b) Identifying possible enhancements of safety and reliability;" *(Note: It refers to the Annex)*	X	*Incorporation of new item (b): Determination of EOPs and other procedures are covered by (a)/(b)*
Observer ENISS-5	1.4	1.4. The objective of this Safety Guide is to provide	Where does the draft describe a method for		*See comment GER-1 above*	X	*About (b): See items (e) to (i)*

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		recommendations and guidance on performing deterministic safety analysis for designers, operators, regulators and technical support organizations. It also provides recommendations on the use of deterministic safety analysis in: (a) Demonstrating or assessing compliance with regulatory requirements; ~~(b) Identifying possible enhancements of safety and reliability~~;	"Identifying possible enhancements of safety and reliability" applying DSA? "Reliability" is beside "effectiveness" one of the most essential characteristic of safety-related SSC´s to realize safety functions at the required level of safety.				*in the Annex*
Observer EC/JRC-1	1.4/1	The objective of this Safety Guide is to provide recommendations and guidance on performing deterministic safety analysis under the objectives established in paragraph 5.75 of SSR-2/1 Rev. 1 and paragraph 4.15 of GSR Part 4 Rev. 1	1. Identification of target actors of deterministic safety analysis falls more within the scope section of the guide. 2. Bullet (a) is embedded in 5.75 (d) of SSR-2/1; bullet (b) is embedded in 4.15 of GSR Part 4 Rev. 1. It is somewhat misleading to set these two objectives aside, moreover since link with previous paragraph in the text is performed through linguistic sentence connector 'also', i.e. as they will go beyond established uses of deterministic safety analysis by the IAEA.			X	*See Germany-1 above. The change of formulation seems not necessary*

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Observer EC/JRC-2	1.5/1	This Safety Guide applies to new and existing nuclear power plants.	1. All mentions made throughout the 'scope' section pointing out at the target facility of the safety guide should be wrapped up. In this sense, 1.5 and first part of 1.6 are brought together into one single para. 2. It is somewhat confusing to lift up only two of the objectives within the wide myriad of objectives pursued through deterministic safety analysis. These two objectives should only be explicitly mentioned as long as the rest of the objectives included in 5.75 of SSR-2/1 falls beyond the scope of the current guide –which is not the case. 2. 1.5 and first sentence of 1.6 should be merged and rephrased.			X	*See USA-1 to §1.6 below*
Observer ENISS-9	1.5…1.14	SCOPE 1.5. This Safety Guide applies to…	The scope of this Guide is unnecessary broadly described and should be significantly shortened (e.g. deletion of 1.11 and 1.12).			X	*See CAN-42 below. To §1.11.* *Section 1 does not provide guidance/ recommendations. §1.11 and §1.12*

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							are included for clarification
Canada 3	1.6 1st sentence	1.6. This Safety Guide focuses primarily on the deterministic safety analysis for the design safety of new[x] nuclear power plants and, as far as reasonably practicable or achievable, also the safety re-evaluation or assessment of existing nuclear power plants when operating organizations review their safety assessment. [footnote x] The meanings of "new" and "existing" and their application are as described in SSR-2/1 paragraphs 1.1 to 1.3.	There can be problems caused by use of terms like "new" or "present NPP' and "existing NPP". The guide must explain the dividing line between new and existing. In particular, we need to lock the definition to the date of publication, otherwise "new" NPPs become "existing" once they enter service and all the requirements become guidance! SSR-2/1 para 1.1 clearly implies that the publication date of a standard is considered "present". SSR-2/1 para 1.2 and 1.3 considers NPPs to be "existing" when they are in operation, or they are under construction, or the design has been approved by regulatory body			X	*Definition of "new" is outside the scope of this Safety Guide and applies to many other. The terms "existing" and "new" are used in the Glossary and in the Safety Requirements.*
USA 1	1.6	The guidance provided is	The standard is not		"The guidance provided is		

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	Second sentence (p. 2)	~~intended to be as much as possible technology neutral, although it is particularly~~ based on experience with deterministic safety analysis for existing water cooled reactors and should be used with caution in considering new water-cooled or other advanced reactor designs.	technology neutral. It clearly applies to current light- and heavy-water-cooled reactors, and it may apply to some new water-cooled designs. It is not clear that it has any relation to gas-cooled or other advanced reactor designs.		intended to be as much as possible consistent with §1.6 of SSR-2/1 (Rev. 1) [1] and ~~technology neutral, although~~ it is particularly based on experience with deterministic safety analysis for water cooled reactors."		
Observer EC/JRC-3	1.6/4	"This Safety Guide addresses the main aspects concerning the performance of deterministic safety analysis for designers, operators, regulators and technical support organizations as listed in paragraph 5.75 of SSR-2/1 Rev. 1, including improvements in safety provisions through backfitting design."	1. One of the most far-reaching consequences of Fukushima Dai-chi lessons learned consists of the installation of totally new safety systems (traditionally binned under the category of 'backfitting') where Deterministic Safety Analysis plays a fundamental role, e.g. for the design phase of related severe-accident mitigating systems such as PARs, FCV, etc. Since this is a sound aspect of deterministic safety analysis, I would outline it explicitly.			X	*The change seems not justified (see other comments to this paragraph)*
Observer EC/JRC-4	1.6/4	Second sentence to be replaced in new dedicated para.	First and second sentence of 1.6 touches different issues: First sentence is			X	*Taking into account the other comments it seems better not*

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			about whether the safety guide applies to new / existing plants while second sentence talks about the type of plant design.				*to split the paragraph*
Japan 1 Line 2	1.8.	radioactive ~~substances~~ materials	To be consistent with used in SSR-2/1 (Rev. 1).	X			
Belgium 1	1.8 and 3.51	Make article 1.8 and articles 3.51 till 3.54 coherent.	At one hand, art. 1.8 says that internal and external hazards are not covered. At the other hand, article 3.51 till 3.54 cover these hazards. This seems not coherent.		*1.8 (second sentence):* … "Analysis of hazards themselves, either internal or external (natural or human induced) is not covered by this Guide, although the effects and loads potentially inducing the failures in plant systems are taken into account in determining initiating events to be analysed~~.~~ *(3.51 is treated with the comments to Section 3)*		
Observer ENISS-6	1.8 Line 1.	This Safety Guide deals with those failures in the reactor core, reactor coolant system (RCS), fuel storage, systems containing radioactive substances or any other system that ~~affect~~ have the potential to challenge performance of safety	In a DSA it is shown, that failures do not affect safety functions.		"…any other system that has the potential to affect the performance of safety functions…"		

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		functions potentially leading to loss of physical barriers against releases of radioactive substances. Analysis of hazards, either internal or external (natural or human induced) is not covered by this Guide, although the loads potentially inducing the failures in plant systems are taken into account in determining initiating events to be analysed~~.~~					
Observer EC/JRC-5	1.8/3	… against releases of radioactive substances in all operational conditions of the plant (i.e. full power, low power and shutdown).	The scope does not say anything about operational conditions of the plant, e.g. low power and shutdown, whereas SSG-3 on PSA indeed does. It is clear that PSA models must be specifically developed to LP&S modes but also emphasis on deterministic safety analysis applied to LP&S should be included in the scope.			X	*It seems understood in the sentence*
Observer ENISS-7	1.9	This Safety Guide is devoted to the deterministic safety analysis for design or licensing purposes, which are aimed at demonstration of compliance with acceptance criteria ~~with adequate~~	Acceptance criteria may already integrate margins with regards to the safety limit.		*Editorial* "… which are aimed at ~~demonstration~~ demonstrating, with adequate margins, ~~of~~ compliance with acceptance criteria ~~with adequate margins~~.		

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		~~margins~~.					
Observer EC/JRC-6	1.9/all	To be removed because of redundancy	First and second sentences embedded in new para 1.6 when objectives are referred to.		*(It seems better to keep it, see ENISS-7 above)*		
Observer EC/JRC-7	1.10/all	This Safety Guide covers different options available for performing deterministic safety analysis, whether conservative or not.	Terminology in Table 2 makes use of terms standing for different options in performing deterministic safety analysis, among which 'conservative' and 'realistic'. If 1.10 employs exactly the same terms, it is not clear whether such options are being referred or if they are being used under their conventional meaning. In fact, para 1.16 on structure of the report, line 13, rather talks about "conservative and best estimate". To avoid misleading, rewording is suggested.	X			
Canada 42	1.11	*Suggest the following changes,* This Safety Guide focuses on neutronic, thermal hydraulic, fuel (and fuel channel for PHWR) and radiological analysis.	The behaviour of fuel (& fuel channel for PHWR) is critical in the evaluation against the acceptance criteria.	X			
Observer	1.12/1	The extent of radiological	Source term release is		"1.12. The extent of radiological		

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EC/JRC-8		analysis in this Safety Guide is limited to the transport and release analysis of radioactive substances within the buildings of the nuclear power plant, in particular in anticipated operational occurrences and accident conditions, as one of the inputs for determining the radiation doses to the nuclear power plant staff (see GSR Part 3). All aspects going beyond the determination of source term release to the environment, such as dose calculation, radioactive gaseous and liquid effluent calculations or dispersion of radioactive substances in the environment, are not covered by this Safety Guide. While general rules… for example in *[5]*.	also comprised within the radiological analysis as accounted for in the current safety guide. Instead of splitting similar intimately related contents between 1.12 and 1.13, it would become better organized if combining them into one single para addressing all aspects related to radiology.		analysis in this Safety Guide include ~~is limited to~~ the transport and release analysis of radioactive substances ~~inside~~ within the buildings and structures of the nuclear power plant, in particular in anticipated operational occurrences and accident conditions, as one of the inputs for determining the radiation doses to the nuclear power plant staff (see GSR Part 3) [4]. The aspects going beyond the determination of source term release to the environment, such as dose calculation, radioactive gaseous and liquid effluent calculations or dispersion of radioactive substances in the environment, are not covered by this Safety Guide. It is however recognized that minimization of the staff…”		
Czech 1	1.13 Last line	Such specific guidance can be found in ~~other~~ IAEA Safety Guide, ~~for example in~~ [5].	When using singular word Guide, wording “for example” doesn’t sense.		“…found in other IAEA Safety Guides, e.g. ~~for example~~ in [5].” *(See ENISS-8 below)*		
Canada 4	1.13, sentences 1 & 2	1.13. This Safety Guide also covers some aspects of the analysis of radiological releases. ~~radiological aspects associated with different plant~~	The first sentence is very unclear and the intended meaning is already covered by the following		1.13. This Safety Guide also covers ~~some~~ aspects of the analysis of radiological releases, ~~radiological aspects associated with different plant states with~~		

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		~~states with potential releases of radioactive substances to the environment as the source term evaluation for determining radiation doses to the public.~~ However, these aspects are only covered ~~up to~~ for the determination of the source term to the environment for AOOs and accident conditions.	text. Simplify the text as indicated.		~~potential releases of radioactive substances to the environment as the source term evaluation for determining radiation doses to the public. However, these aspects are only covered~~ up to the determination of the source term to the environment for AOO and accident conditions (§2.16 to §2.18).		
Observer ENISS-8	1.13	While general rules …such analysis. Such specific guidance can be found in other IAEA Safety Guides ~~, for example in [5]~~.	Ref [5] is under revision (revises NS-G-3.2), and the changes introduced are not known. Therefore it's preferable to not give it as an example, or refer to the current published version.		*(See Czech-1 above)* *Editorial clarification: DS427 is indicated provisionally in [5]. The draft is in step 11 and its publication is expected by the time of starting the publication process of DS491. Otherwise NS-G-3.2 would be referenced.*		
Canada 5	1.16 all	Use bullets for each section	This paragraph would be much easier to read if a bullet were used for each section.			X	*Formatting is indicated in IAEA Guidelines (SPESS C)*
USA 2	1.16 (p. 4), Last sentence	Some terms and explanations for ~~consideration in the preparation or revision of safety standards and so for~~ possible inclusion in the IAEA Safety Glossary are provided at the end, under Definitions. These terms and explanations	Current wording of sentence is confusing.		*Last sentence will be removed*		

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		should be considered in the preparation or revision of safety standards.					
Observer ENISS-10	1.16	Besides this introduction, this Safety Guide consists of ~~nine~~ **eight** additional sections and one annex.	The SG has 9 sections in totality (8 in addition to the introduction).	X	*Editorial*		

Section 2
DS491 Step 7: Deterministic Safety Analysis for NPPs

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Korea 1 (Rev 1)	General to Section 2	[general comment] Complementary relation between DSA and PSA should be briefly described in the Chapter 2 General Considerations. (GSR Part 4 para 4.53 and SSR-2/1 requirement 10)	[general comment]			**X**	*Section 2 has explanatory nature and does not provide recommendations to meet requirements. On the other hand, clarifications about the complementarity or recommendations to meet the requirements seems not necessary under the scope of this Safety Guide*
Korea 2 (Rev 1)	General to Section 2	[general comment] It could be useful to provide a flowchart of the basic steps in the safety analysis procedures in the Chapter 2 General Considerations. An example is shown in the FIG.I-1 of Annex 1, Safety Report Series	[general comment]			X	*Out of the scope of this Safety Guide. It may be more commonly included in safety reports or similar documents*

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		No.23					
Korea 3 (Rev 1)	General to Section 2	[general comment] It may be necessary to describe, in general, the management of the safety analysis required in GSR Part 4 Requirement 22 ("The process by which the safety assessment is produced shall be planned, organized, applied, audited and reviewed.") in the Chapter 2 or Chapter 8 of DS491.	[general comment]		*(See current §3.1 about "Management System")*	***X***	*This Safety Guide deals with Deterministic Safety Analysis. Safety Analysis in general and Safety Assessment are out of its scope*
Czech 2	2.1 Line 5	"…Deterministic safety analysis, supplemented by a number of investigations such as those related to fabrication, testing, inspection, evaluation of the operating experience and by PSA, is also aimed to contribute to demonstrate that the source term and eventually radiological consequences of different plant states are acceptable and that early ~~or~~ large releases are practically eliminated."	The past radiation emergencies (Chernobyl and Fukushima Daichi) demonstrate that large releases are not practically eliminated. "But early large releases" can be eliminated. See text in para 3.25 and 3.55 and others of this guide too. In some para text "large or early" instead of "early or large" is used. What are the differences? We can compare early to late or large to small, but compare early to large seems to be strange. These things are two different categories.		*According to the wording used in §2.13 (4) and §5.31 of SSR-2/1 (Rev. 1):* "…different plant states are acceptable and that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release~~s~~ are 'practically eliminated'."		

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Ukraine 1	2.1 last line and Para 2.18 (b)	*[re 2.1 last line]* To add "radioactive" before "releases". "… is also aimed to contribute to demonstrate that the source term and eventually radiological consequences of different plant states are acceptable and that early or large radioactive releases are practically eliminated".	To specify the formulation.		*(2.1: See Czech-2 above)* *(2.18 (b) will be also updated accordingly; see comments below about this paragraph;)*		
France 1	2.1 Line 5	Deterministic safety analysis, supplemented by a number of investigations such as those related to fabrication, testing, inspection, evaluation of the operating experience and by PSA, is also aimed to contribute to demonstrate that the source term and eventually radiological consequences of different plant states are acceptable and that situations which could lead to early or large releases are practically eliminated.	The "practical elimination" approach should be related to accidental situations or conditions or sequences and not to releases : consistency with SSR-2/1 §2.11, 4.3, 5.31 and INSAG 10 §5.1.		*(2.1: See Czech-2 above)*		
Observer ENISS-11	2.1 Lines 1-3	2.1 The objective of deterministic safety analysis for nuclear power plants is to confirm that safety functions and the needed ~~plant systems~~ SSCs, in combination where	Objective of deterministic safety analysis (DSA) is exclusively focused on sufficient "effectiveness" of the safety functions and their related SSCs in		2.1 The objective of deterministic safety analysis for nuclear power plants is to confirm that the safety functions and the needed ~~plant systems~~ SSCs, in combination where		

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		relevant with operator actions, are capable and sufficiently effective, with adequate safety margins, to keep the radiological releases from the plant ~~under~~ within acceptable limits.	contrast to the objective of a probabilistic analysis where the "reliability" of SSCs and safety functions are primarily in the focus. Exchanging "under" by "within" is suggested to correct English and even to be factual right. Otherwise it could be misinterpreted as rad. releases that have to be kept below the accepted release interval which is above operational release values but below assumed accident values and shortly circumscribed by "acceptable limits".		relevant with operator actions, are capable and ~~sufficient~~ sufficiently effective, with adequate safety margins, to keep the radiological releases from the plant ~~under~~ within acceptable limits.		
Observer ENISS-12	2.1 Line 4	Deterministic safety analysis is aimed to demonstrate that SSCs designed as active or passive barriers to the release of radioactive material from the plant will maintain their integrity and function to the extent required.	For more clarity and precise expression	X			
Observer ENISS-13	2.1	Deterministic safety analysis, supplemented by further specific information and analysis ~~a number of investigations~~ such as those related to fabrication, testing, inspection, evaluation of the	Complement to clearly state that practical elimination is associated to situations with core melt (see WENRA Safety of new NPP designs). Alternatively, in order to	X *(First modification)*	*Second modification: Covered in Czech-2 above*		

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		operating experience and by PSA, is also aimed to contribute to demonstrate that the source term and eventually radiological consequences of different plant states are acceptable and that accidents with core melt which would lead to early or large releases are practically eliminated.	align with SSR-2/1, "early or" should be omitted.				
Observer EC/JRC-9	2.1 and 2.2 / all	The objectives of deterministic safety analysis are those found in para. 5.75 of SSR-2/1 Rev. 11 and paragraph 4.15 of GSR Part 4 Rev. 1.	Objectives are listed in a very clear manner in overarching SSR-2/1 Rev. 1 guide. For the sake of clarity and to avoid misleading, they should be reproduced here without modifications. If desired, only further explanations of each of them might be added. For instance, first sentence talks about "the objective" when actually there is more than one objective; besides, it says that the objective is "… to confirm that plant systems, in combination where relevant with operator actions…". However, LBLOCA containment peak pressure in critical flow conditions –checked with deterministic safety analysis– right after the			X	*(See ENISS-11 above). This formulation is not used in the Safety Guide*

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			break looks at containment pressure design so that safety systems capability to withstand such peak does not apply in this context (instead, passive heat sinks play a fundamental role if best-estimate calculations are to be performed). Another example is the use of such analysis in meeting with operational limits and conditions (i.e. Technical Specifications) where mentioned objectives in paras 2.1 and 2.2 do not match suitably.				
Observer EC/JRC-10	2.3/5	Computational simulations should be carried out specifically for all operational conditions of the plant from full power to shutdown.	It should be strongly stated the need for building as many input models of the plant as operational states exist.		*("Should" statements are not used in Section2, see 1.12 line 2)* "… ~~The computations~~ Computational simulations are ~~should be~~ carried out specifically for predetermined operating modes and plant states ~~configurations"~~		
Germany 2	2.4	2.4. The results of computations are spatial and time dependent values of various physical variables (e.g. neutron flux; thermal power of the reactor; pressures, temperatures, flow rates and velocit~~y~~ies of the	In principle the concentrations of combustible gases like hydrogen and carbon monoxide are interesting. Thus, limitation of the concentrations to combustible gases.	X			

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		primary coolant; loads to physical barriers; concentrations of combustible gases, …	The physical properties of the primary coolant have to be checked at different locations of the reactor circuit.				
Canada 6	2.5, 1st sentence	Add new first sentence: Acceptance criteria are essential components of deterministic safety analysis, since they are used for judgment of acceptability of the demonstration of safety of a nuclear power plant.	Some introductory text would make this paragraph clearer. The suggested text was originally at the end of the preceding paragraph, deleted during internal IAEA review.		*At the beginning of 2.5 it will be added:* "Acceptance criteria are used in deterministic safety analysis for judgment of acceptability of the demonstration of safety of a nuclear power plant. The acceptance criteria can be		
Czech 3	2.5 Line 7	…..these are criteria either directly related to the consequences of operational states or accident conditions or to the integrity of barriers against releases of radiation exposure and radioactive materials	Physical barriers serve not only against releases of radioactive material but against radiation too.			X	*Radiation exposure is out of the scope of this Safety Guide (and is not a safety criterion for DSA).*
Canada 7	2.6 (All)	Delete paragraph 2.6 and change all occurrences of "safety criteria" to "acceptance criteria". There are two in para 2.5 and one in para 7.21.	The purpose of the paragraph appears to be to explain that "safety criteria" are "acceptance criteria", but the text is very unclear. Since the only occurrences of "safety criteria" are in paragraphs 2.5 and 7.21, it would be much simpler to change those occurrences		*About 7.21:* *In 7.21 "*safety criteria*" will be replaced by "*acceptance criteria*".* *(See also ENISS-14, below)*	X	*About deleting 2.6: Para 2.6 defines acceptance criteria to be equal to safety criteria.*

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			to "acceptance criteria".				
Observer ENISS-14	2.6	In this Safety Guide, only the ~~safety~~ acceptance criteria that are the targets of deterministic safety analysis are addressed ~~used in connection with the deterministic safety analysis and the wording acceptance criteria then refers to safety criteria~~. These acceptance criteria may include decoupling margins with respect to safety criteria.	In this para., safety criteria and acceptance criteria are merged. We strongly insist on the fact that acceptance criteria shall not systematically be mixed with safety criteria. In some cases, for convenience, acceptance criteria may be defined to include decoupling margins with respect to the safety criteria. As an example, one can choose to adopt a "no core uncovery" acceptance criteria in case of LOCA whereas the safety criteria shall rather adress the cladding embrittlement, the hydrogen production…		"In this Safety Guide, only the ~~safety~~ acceptance criteria that are the targets of deterministic safety analysis are addressed ~~used in connection with the deterministic safety analysis and the wording acceptance criteria then refers to safety criteria~~. The regulatory body may decide to approve acceptance criteria that may include margins with respect to safety criteria.		
Japan 2	2.7	Several methods for performing uncertainty analysis have been published (e.g. in Safety Report Series No. 52 [10] ~~para 6.21-6.29 and 7.43~~).	Para 6.24-6.29 and 7.43 do not exist in the referenced document [10].		"2.7. In this Safety Guide, uncertainty analysis are addressed in §6.21-§6.29. Several methods for performing uncertainty analysis have been published (e.g. in Safety Report Series No. 52 [10] ~~para 6.21-6.29 and 7.43~~).		
Observer ENISS-15	2.7/ after last line	The assessment of uncertainty is fit for purpose in the safety analysis, according to an	For clarification, and to allow combination of the methods identified.		*See Japan-2 above*	X	*Clarification / detail seems not necessary*

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		appropriate method. Several methods for performing uncertainty analysis …. They include: (a) Use of …; (b) Use of …; (c) Use of …. A combination of (a), (b), and (c) is also possible.					
Observer EC/JRC-11	2.7/2	Related to reference as indicated in the text: Safety Report Series No. 52 [10] para 6.21 – 6.29 and 7.43	Paras in the referenced report are not numbered, i.e. it does not exist para 6.21. Please correct.		*(See Japan-2 above)*		
Observer EC/JRC-12	2.7/All	Additional information on uncertainty analysis should be included	Even if not aimed at exhaustively describe uncertainty analysis main steps and sound methods, the information provided here is too poorly described and should be extended, at least, to touch fundamental aspects just equivalently to what done in other introductory sections under point 2 on "general considerations".			X	*No specific suggestion provided*
Canada 8	2.8	Correct Table 2 to Table 1.		X			
Korea	2.8	Table ~~2~~ Table 1	Errata	X			
Observer ENISS-16	2.8/Table 2, first line	Replace ~~“type of initial~~…” by “other initial…”	“systems availability” is part of “initial and boundary conditions”			X	*Systems availability can be considered as part of methodology and not an initial or*

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							boundary condition
Observer ENISS-17	§2.8/Table 2	Add a line in Table 2 to address DEC practices: An intermediate case between the BEPU and the realistic approach should be mentioned, where the assumption on systems availability would be "Best Estimate", rather than "conservative"	As mentioned in §7.50, the "Single Failure" rule shall not be applied in the frame of Design Extension Conditions. The case of systems availability during preventive maintenance is not explicitly treated in §7 but could be considered as very penalizing regarding the low initiating event frequency associated to this category of events.			X	*Major change. The main options currently used are included. The suggestion could add confusion regarding the differences with existing options 3 and 4.*
Observer EC/JRC-13	2.9/5	In a conservative approach, evaluation models for phenomena simulation implemented into the codes deterministically lead to unfavorable effects regarding specific acceptance criteria calculation. Furthermore, this approach is also based on selecting scenario initial and boundary conditions increasing mass and energy loads challenging safety systems and radiological barriers. Nonetheless, since this approach does not provide with the actual safety margins (Bucalossi, 2008)[1], and since	1. Current sentence presents unclear wording. 2. It does not properly distinguish between evaluation models and boundary and initial conditions. 3. It is relevant to bring here the (IAEA, 2008) statement about this full conservative approach.			X	*The suggestion could be considered as too detailed*

[1] Bucalossi A., "current use of best estimate plus uncertainty methods on operational procedures addressing normal and emergency conditions", European Commission Joint Research Centre Technical Report, 2008

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		there are scenarios where the real value of the relevant plant parameter provided by the calculation of the code is unknown –due to the deliberate pessimistic criteria characterizing the evaluation models–, sometimes even leading to non-conservative results (D'Auria et al., 2006)2, the use of this approach is no longer recommended by (IAEA, 2008)3.					
Canada 43	2.10 First sentence	Suggest the following changes, *At present experimental research has resulted in a significant increase of knowledge and the development of computer codes has improved the ability to achieve calculated results that correspond more accurately to experimental results and post-accident conditions in power plant*	Although it is important for the computer code to accurately reproduce experimental results, it is post-accident plant conditions that are ultimately of interests.		"…to experimental results and recorded event sequences in nuclear power plants. "		
Observer EC/JRC-14	2.10/1	At present, the state of the art of phenomena taking place in plant states from normal	1. Rephrasing of para 2.10 aims, on one hand, at limiting this increase		*(See Canada-43 above)*		

[2] D'Auria F. Bousbia Salah A. Petruzzi A., Del Nevo A., "State of the art in using best estimate calculation tools in nuclear technology", Nuclear Engineering and Technology, Vol. 38, No. 1, 2006
[3] International Atomic Energy Agency, "Best Estimate Safety Analysis for Nuclear Power Plants: Uncertainty Evaluation", Safety Report Series No. 52, 2008

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		operation up to design basis accidents has significantly improved as a result of wider and more reliable experimental research. Benefits from this increase in knowledge have subsequently been translated into corresponding improvement in simulation codes.	of knowledge to and up the DBA field, and on the other, removing / replacing / reformulating drawbacks in previous para 2.9.				
Canada 44	2.11 Second sentence	Suggest the following changes, *Best estimate codes are used in combination with conservative initial and boundary conditions, as well as with conservative assumptions regarding the availability of systems, assuming that all uncertainties associated with the code models are well established and plant parameters are ~~bounded~~ conservative based on plant operating experience.*	It is important for the plant parameters to be conservative, not necessarily bounded. With respect to code model uncertainties, the requirements should be well established, and not bounded.	X	are well established and plant parameters are ~~bounded~~ conservative based on plant operating experience.		
Observer EC/JRC-15	2.11/5	First part of next-to-last sentence (The complete analysis…) to be removed.	Computer code validation should be requested in all options so no reason to state it here linked to option 2.		"…The complete analysis requires ~~adequate validation of the computer code and~~ use of sensitivity studies to justify conservative selection of input data"		

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Observer EC/JRC-16	2.11/6	Option 2 is commonly used for DBA and ~~conservative~~ anticipated operational occurrence analysis yet some national regulations, such as US Code of Federal Regulations, does not permit option 2, while allowing applying either option 1 and 3.	If current para 2.11 includes arguments on the practical use of deterministic safety analysis applications, it would be significant to balance the current statement ("commonly used") by introducing sound exceptions to avoid readers wrong belief in making 'common' a sort of equivalence to 'consensus'.		*First part: See France-2 below to 2.12* … Option 2 is commonly used for DBA and for conservative analysis of anticipated operational occurrences ~~analysis~~ (~~e.g. para6.12~~)." *(see EC/JRC-17)*		*Last part: Such detail seems not necessary*
Observer EC/JRC-17	2.11/7	Reference into brackets to para 6.12 deals with option 2 further description so it should be removed or replaced above at the beginning of para 2.11 when option 2 is first mentioned	This reference is unnecessary. References within the text should be placed at the first time when they are introduced. If this reference is to be kept, why then not applying the same for option 1 when introduced in para 2.9?		"2.11. Option 2 is a combined approach based on the use of 'best estimate' models and computer codes instead of conservative ones (§6.12). ...		
Observer EC/JRC-18	2.12/2	… together with as-built plant boundary and initial conditions accounting only for existing uncertainties hence avoiding imposing any deterministic conservative burden.	Dealing with boundary and initial conditions, 'partially most unfavourable' statement is highly ambiguous.			X	*Too detailed*

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Observer EC/JRC-19	2.12/3	In turn, avoidance of all type of conservatisms requires comprehensive analysis of the entire spectrum of uncertainty sources when simulating plant state scenarios to ensure success in mitigating systems performance and radiological barriers.	The meaning of 'the conservatism required in analysis of DBAs' is unclear. Which regulation is requiring it? Besides, removing conservatisms in performing safety analysis is precisely the goal pursued when switching from option 1 to 4, where option 4 is free of any degree of imposed conservatism.			X	*Too detailed*
France 2	2.12 Line 5	"… Option 3 contains a certain level of conservatism and is at present accepted for some DBA and ~~conservative~~ anticipated operational occurrences analyses (e.g. para 6.21).	Word 'conservative' before AOO is to be removed as already mentioned at the beginning of the sentence.		"… Option 3 contains a certain level of conservatism and is at present accepted for some DBA and for conservative analyses of anticipated operational occurrences ~~analyses (e.g. para 6.21)."~~		
Observer EC/JRC-20	2.12/5	Option 3 contains limited degree of conservatism only related to boundary and initial conditions and is at present accepted in some national regulations for DBA and anticipated operational occurrence analysis.	It seems that wording 'some' applies to 'DBA and anticipated operational occurrence'. However, this is wrong since regulation likely focuses on deterministic safety analysis applied to an entire set of so-called plant states so that if one specific option, e.g. option 3, can be applied to LBLOCA, it will		*(See France-2 above)*		

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			certainly be applicable – following that same regulation– to a SGTR. Therefore I believe 'some' applies to national regulations giving utilities and TSO the possibility of applying option 3 in this field.				
Observer EC/JRC-21	2.12/7	Last sentence should be removed	There is no need for explicitly mentioning one of the crucial aspects related to a correct uncertainty analysis in option 3. There are also several other ones, e.g. selection of significant and high-uncertainty phenomena, identification of user-effect sources of uncertainty (including nodalization analysis), which also have a very important role in properly conducting the uncertainty analysis.	XEC/JRC-21			
Observer EC/JRC-22	2.13/All	Removed the entire para	This para does not contain any added value. Besides, it does not correspond to reality when stating that availability of extensive data is associated to best-			X	

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			estimate boundary and initial condition approach. If this were the case, best estimate in boundary and initial conditions is related to option 3, hence uncertainty analysis will be mandatory. On the other hand, utilities and TSO in charge of performing such deterministic analysis have access to as built, extensive and detailed data of the plant. Therefore and according to these two arguments, option 3 should be the first option for utilities and TSOs. Nonetheless, most applications worldwide still make use of option 2.				
USA 3	2.15 Line 2	…Option 4 may be appropriate for realistic analysis of anticipated operational occurrences aimed at assessment of control system capability and in general for best estimate design extension conditions analysis (see paras 7.17 and 7.54). Safety assessments	Safety assessments of operating events that may require short term relaxation of regulatory requirements are another potential application for best estimate modelling.	X	"… Option 4 may be appropriate for realistic analysis of anticipated operational occurrences (§7.17-§7.54) aimed at assessment of control system capability and in general for best estimate analysis design extension conditions analysis (§7.45-§7.67~~see paras 7.17 and 7.54~~). Additionally, this option		

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		performed for operating events that may require short term relaxation of regulatory requirements may rely on best estimate modelling. More detailed information…			may be used ~~Safety assessments performed~~ for the analysis of operating events that may require short term relaxation of regulatory requirements ~~may rely on best estimate modelling~~. More detailed information…		
Japan 3	2.15 Line 4	... More detailed information regarding modelling assumptions applicable for different options is provided in ~~section 8~~ section 7 of this Safety Guide.	Editorial.	X			
Korea	2.15 Line 4	More detailed information regarding modelling assumptions applicable for different options is provided in section ~~8~~ 7 of this Safety Guide.	Errata	X			
Observer EC/JRC-23	2.15/1	Option 4 allows using best estimate code modelling, system availability assumptions and initial and boundary conditions.	Ambiguous sentence when referring to parameters, on one hand, and modelling, on the other.	X			
Observer EC/JRC-24	2.15/3	… aimed at assessment of control system capability (see paras 7.17 and 7.54).	The fact of removing last part of second sentence in para 2.15 stems from the increasing awareness on the strong impact that uncertainties have in the field of severe accidents. This issue will be	X			

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			developed later on in comment XXX				
Czech 4	2.16	In accordance with Ref. [3] (IAEA Safety Glossary) the source term is ’The amount and isotopic composition of material released (or postulated to be released) from the facility’; it is ’used in modelling releases of radionuclides………	This definition is missing the timing of the radioactive substances releases. Modification needed. This definition speaks only about the fraction of the fission products released from the core or from any other source at NPP.			X	*The definition used in this Safety Guide has to be the one of the IAEA Safety Glossary*
France 3	2.16 Line 1	Deterministic safety analysis includes ~~as its essential component~~ determination of the source	‘as its essential component’ to be removed as DSA includes several other essential components		“2.16 Deterministic safety analysis includes as one of its essential components determination of the source…”		
Observer EC/JRC-25	2.16/1	One of the sound results potentially drawn from deterministic safety analysis is source term calculation, which will ultimately serve for prediction of dispersion of radioactive substances to the environment and eventually does to the plant staff, to the public and radiological impact on the environment.	I don't agree when saying that source term determination is the essential component of deterministic safety analysis. Deterministic safety analysis have a wide spectrum of objectives each of which can strongly impact on safety analysis and assessment activities. Just to make an example, source term categorization as a consequence of		*(See France-3 above)*		

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			severe-accident sequence simulation with safety codes is performed through Level 2 PRA tool. However, up-to-date significance and number of consequences from Level 1 PRA application highly exceeds those coming from Level 2 PRA. Moreover, core damage figure of merit have also a much stronger impact within FSAR than source term categorization.				
Observer EC/JRC-26	2.17/All	Source term evaluation under accident conditions requires simulation code capabilities dealing with fission product release from fuel elements, transport through primary system and containment or spent fuel pool building, and related chemistry. Risk-dominant and earliest, largest sequences leading to source term release to outside containment / spent fuel pool building environment or attached buildings should be taken into account.	It is unclear which actor is responsible for tasks identified in para 2.17. Safety engineer in charge of carrying out deterministic safety analysis will calculate source term by making use of dedicated simulation code. Therefore, I would recommend to reorient para 2.17 towards code capabilities in terms of affected source term phenomena.		First sentence: "Under accident conditions, source term evaluation requires simulation code capabilities dealing with fission product release from fuel elements, transport through primary system and containment or spent fuel pool building~~,~~ and related chemistry"		*Second sentence would represent too much detail*
Germany 3	2.18	2.18. Source term is evaluated for operational states and accident conditions for the	The demonstration that early or large releases can be excluded can only be		"… (b) To support ~~by means of its quantification~~ the demonstration		

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		following reasons: (a) To ensure that the design is optimized so that the source term will be reduced to a level that is as low as reasonably achievable in all plant states; (b) To support by means of its quantification the demonstration that early or large releases can be considered as practically eliminated (should be done in co-operation with supporting probabilistic safety analyses); (c) To demonstrate that the design ensures that requirements for radiation protection, including restrictions on doses, are met; (d) To provide a basis for the emergency arrangements[2] that are required to protect human life, health, property and the environment in case of an emergency at the nuclear power plant; (e) To specify the conditions for the qualification of the equipment required to withstand accident conditions. (f) Provision of databases for training activities regarding emergency preparedness. (g) Supporting Level 2 PSA	done in co-operation with probabilistc safety analyses. Thus, the usage of only deterministic event analyses is not sufficient. Other relevant objectives of source term analyses are to deliver data for the training of emergency preparedness and supporting Level 2 PSA analyses.		that … *(Regarding the last part of this bullet, see resolution to comment Czech-2 above)* … (f) To provide data for training activities regarding emergency arrangements.		

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		analyses.					
Czech 5	2.18 Bullet (b)	To support by means of its quantification the demonstration that early ~~or~~ large releases can be considered as practically eliminated;	See comment 2.		*First part: See resolution to Germany-3 above.* *Second part: See resolution to Czech-2 above (para 2.1)*		
Ukraine	2.18 Bullet (b)	To add "*radioactive"* before "releases". "… is also aimed to contribute to demonstrate that the source term and eventually radiological consequences of different plant states are acceptable and that early or large ***radioactive*** releases are practically eliminated".	To specify the formulation.		*See resolution to Czech-2 above (para 2.1)*		
France 4	2.18 Bullet (b)	~~(b) To support by means of its quantification the demonstration that early or large releases can be considered as practically eliminated~~	1) We are not sure to understand this sentence. It seems to be in contradiction with 3.57 : "Consequences of event sequences that have been 'practically eliminated' do not need themselves to be deterministically analysed…." 2) Moreover, as for the previous comment, the "practical elimination" approach should be related to accidental situations and		*See resolution to Germany above, regarding §2.18 (b)*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			not to releases – see comment 1				
Observer EC/JRC-27	2.18/(b)	*Footnote 6 should be replaced / added here in* 'practically eliminated'	'Practically eliminated' statement appears here for the first time so corresponding clarification note should be included.		*See Germany-3*	X	*It appears also in 2.1. The footnote is placed in a "should" statement*
Canada 45	2.18	Suggest a note be added to explain that thermal hydraulic conditions are equally important as the source term for equipment qualification.	Qualification of equipment is required to withstand the source term and thermal hydraulic accident conditions			X	*The subsection covers source term.*
Observer EC/JRC-28	2.18/(c)	First sentence of footnote 2 should be added here.	Last sentence of para 1.12 says that 'determination of the doses to the nuclear power plant staff is therefore not covered by this Safety Guide", hence footnote 2 on indicating that this reason goes beyond this Safety Guide should also apply here.			X	*Preferable not to enter into that detail in (c)*
Observer EC/JRC-29	2.19/new (2.18??)	To include a new bullet (f) such (f) To characterize so-called Level 2 PRA Release Categories and quantify related figures of merit, e.g. LERF.	Unless it is explicitly stated that the listed reasons only affect deterministic safety analysis, Level 2 PRA results on Release Categories constitute a key aspect of safety analysis interacting with deterministic safety analysis by making use of		*(It seems applicable to 2.18 instead of 2.19)* *See Germany-3*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			simulation code outcomes which should be here taken into account.				
Observer EC/JRC-30	2.19/new (2.18??)	To include a new bullet (g) such (g) To help with the engineering design process related to severe-accident mitigating systems such as Filtered Containment Venting.	Decision criteria on some of the backfitting systems may include minimization of source term release and associated transport heat outside containment.	X	*(It seems applicable to 2.18 instead of 2.19)* (g) To support safety design of mitigating systems related to severe-accident (e.g. Filtered Containment Venting)		

Section 3
DS491 Step 7: Deterministic Safety Analysis for NPPs

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Pakistan 1	3/	Table 1 of SSG-2 (2009) may be included by modifying categorization of plant states according to the definition of the plant states depicted in figure at page 65 of SSR-2/1(rev 1).	In the latest version of SSR 2/1 the Design Extension Conditions (DECs) are introduced and classified as "without significant fuel degradation" and "with core melting". Therefore, the current guide should explain the philosophy of treating each category of DECs during the design process of NPPs.		*Now §3.1* "3.1 In accordance with the definition of "plant states (considered in the design)" from SSR-2/1 (Rev. 1), page 65 [1], the plant states considered in the deterministic … *(§3.2 became §3.1; internal policy indicates not to duplicate)*		
Pakistan 5	3/	Table-2 of SSG-2 (2009) for possible subdivisions of PIEs (AOOs, DBAs and DECs) may be added by including DECs (without significant core melt and with core melting) in section 3 of DS-491.	In order to better explain the subdivisions of PIEs according to the new terminology.			X	*Table 2 of SSG2 (2009) seems outdated; it has been replaced by the data incorporated in § 3.26*

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Observer EC/JRC-31	”POSTULATED INITIATING EVENTS” (3rd Subsection)	N/A	Generic, well-ordered indications aimed at facilitating the design of a structured path to identify and classify PIEs would be an asset. For instance, indications to classify PIEs			X	*More adequate in a lower level document (e.g. Safety Report)*
Canada 9	3.1	Move para 3.1 and its heading “MANAGEMENT SYSTEM” to follow para 2.4 (or somewhere else in section 2).	A paragraph giving the requirement to follow the management system does not belong in a section on Identification and Categorization of PIEs. The management system applies to all of safety analysis and so this paragraph should be in section 2 somewhere under General Considerations.		*Section 2 has descriptive nature and does not include recommendations (no “should” statements.* MANAGEMENT SYSTEM *and §3.1 are moved down to §3.8.*		
Canada 67	3.3	The deterministic safety analysis should consider ~~the~~ postulated initiating events (PIEs) ~~originated~~ originating in any part of the plant that could potentially lead to a radioactive release to the environment in case of failures taking into account ~~requesting the~~ actuation of the control and limitation systems[3] ~~as well as the~~and associated safety functions. ~~and potentially leading to a radioactive release to the environment in case of failures~~. This includes events that can lead to a release of radioactivity not only from the reactor core but from	Grammatically, the sentence as written is awkward and difficult to interpret. For example, PIEs do not request actuation of control and limitation systems.		*Now §3.2 (see CAN-9):* 3.2. The deterministic safety analysis should consider the postulated initiating events (PIEs) originated in any part of the plant ~~and~~ potentially leading to a radioactive release to the environment, with consideration also of additional failures, for example in the control and		

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		other relevant sources such as fuel elements stored at the plant and systems dealing with radioactive materials.			limitation systems[3] and the associated safety functions.		
Observer ENISS-18	3.3	The deterministic safety analysis should consider the postulated initiating events (PIEs) originated in any part of the plant that could potentially lead to abnormal radioactive releases to the environment if unmitigated. ~~requesting the actuation of the control and limitation systems as well as safety functions and potentially leading to a radioactive release to the environment in case of failures~~ This includes events that can lead to a release of radioactivity not only from the reactor core but from other relevant sources such as fuel elements stored at the plant and systems dealing with radioactive materials. For these events, design features such as control and limitations systems and safety systems are implemented so that radioactive releases are kept within acceptable limits.	PIEs should be identified because of their potential abnormal radioactive releases if unmitigated. Then, control & limitation or safety features are implemented to ensure appropriate mitigation.		*See Canada-67 above*		*DSA include normal operation where there are no abnormal releases*
Ukraine 2	Para 3.4.	"3.4. Where applicable, interactions between all reactors, spent fuel storages and any other sources of potential radioactive releases on the given site should be taken into account (***SSR 2/1, § 5.32?***)". Para 5.32 SSR 2/1 deals with combinations of events and failures.	The wrong reference. Moreover the guide itself does not include the explanation how these interactions should be considered in DSA.		*Now §3.3:* 3.3. Where applicable, it should be considered that a single cause can simultaneously initiate PIEs in all reactors, spent fuel storages and any other sources of		

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		The reference is to be checked.			potential radioactive releases on the given site (SSR 2/1 (Rev. 1), § 5.15B) [1]. In case of SSCs important to safety ~~are~~ shared between different units, it should be demonstrated ~~proved~~ that they have sufficient capacity to perform their safety functions as expected.		
Egypt 1	Para 3.4 page 9	Where applicable , interactions between all reactor events and failures , spent fuel storages and any other sources of potential radioactive releases on the given site should be taken into account (SSR 2/1 , & 5.32) [1]	In Para 3.4 ….interaction between all reactors, ….the meaning is not clear for interaction between all reactors and para 5.32 of SSR 2/1 deals with combinations of events and failure.		*See Ukraine-2*		
France 5	3.5	The deterministic safety analysis should be performed for PIEs that can occur in all planned ~~modes~~ conditions or transients of the plant during normal operation at full power and low power, including operation during shutdown.	Planned modes of the plant is not clear;			X	*Mode of operation is used in the IAEA Safety Glossary*
Egypt 2	Para 3.5, page 9 Line 2	…., including operation ~~during~~ and shutdown	,….including operation during shutdown at para 3.5 can be changed to including operation and shutdown or maintenance during shutdown.			X	*It means to include operation in shutdown mode*

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Observer EC/JRC-32	3.5 Line 1	… in all ~~planned modes~~ operational states ~~of the plant~~ during normal operation…	PIEs should be operational-state specific rather than plant-mode specific since every mode can contain several plant configurations each of which greatly different among them in terms of alignment and automatic system availability. This comment should be extended to whenever the text refers to plant modes. Moreover, operational state, or plant operating state, belongs to standard IAEA terminology.			X	*Operational states include both normal operation and AOO. (SSR-2/1 (Rev. 1))*
Germany 4	3.7	3.7. For PIEs initiated in the spent fuel pool, specific operating modes related to typical loadings and fuel handling (e.g. emergency core unloading) should be also considered.	The typical loadings of spent fuel pools (normal loading during power operation, partial loading during overall maintenance inspection, and full loading during repair actions inside RPV, in-service inspections of isolation valves of the reactor circuit and pressure tests) should be mentioned also.		*Now §3.6* 3.6. For PIEs initiated in the spent fuel pool, specific operating modes related to fuel handling and storage ~~(e.g. emergency core unloading)~~ should be also considered.		
France 6	3.8	PIEs potentially taking place during plant operating ~~modes~~ conditions with negligible duration in time may not be considered after careful analysis and assessment of the potential contribution to that sequences, conditions or severe accidents leading	See comment 1 and 5 (for modes)		*(See also comment EC/JRC-33 below)* *Now §3.7* 3.7. PIEs potentially taking place during plant operating modes with negligible		*"operating modes: See France-5 above*

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		to early or large releases.			duration in time may not be considered after careful analysis and quantitative assessment of its potential of contribution to overall risk, including to conditions arising that could lead to an early radioactive release or a large radioactive release.		
Observer EC/JRC-33	3.8 Line 2	… with negligible duration in time may not be considered after careful analysis and **quantitative** assessment of the potential contribution to **overall risk figures of merit**.	First on 'quantitative': real contribution of operational states to risk might be subjectively masked by the relatively short duration of the operational state. In order to suitably weight and potentially neglect one particular operational state, risk should be calculated since it will take into consideration not only time but also the probability of violating safety criteria. Second on 'overall risk': if referred to source term releases, the focus should not only point at large or early releases but to the entire contribution to source term releases. For instance, Fukushima Unit 3 might not		*See resolution in France-6 above*		*Some details are out of the scope of the Safety Guide*

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			be classified as early release yet it should be taken into account when neglecting severe-accident (i.e. DEC) sequences. Third on 'figures of merit': consideration of PIEs should not be assessed only taking associated derived source term releases as safety criteria but all risk-related figures of merit, e.g. impact on core damage, which can significantly differ from consequences on the source term.				
Observer EC/JRC-34	*New*	PIEs identification and classification should be based on similar jeopardized critical safety functions leading to similar safety systems requirements.	Para addressing PIE identification and classification is currently missing. More emphasis and clarification should be made in this regard, for instance, by relocating para 3.30 up to 3.23 or even to the general previous section on 'POSTULATED INITIATING EVENTS'.		*See Germany-8 about §3.23 below (§3.23 and §3.30 have been combined)*		
Germany 5	3.9	(h) …; (i) Normal operation of the spent fuel pool <u>(normal loading during power operation, partial loading during overall maintenance inspection, and full loading during repair actions inside RPV, in-service inspections of isolation valves of the reactor circuit</u>	See comment 4 above		(i) Normal operation modes of the spent fuel pool		

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		and pressure tests); (j) … .					
France 7	3.10	It should be taken into account that in some cases during normal operation, the main plant parameters are changing due to the transfer to different plant ~~modes~~ conditions or the changes in the plant	See comment 5			X	*See resolution to France 5 above*
Germany 6	3.11 Line 3	3.11. Prediction of the plant behaviour in plant states other than normal operation (anticipated operational occurrences, design basis accidents and design extension conditions) should be based on a plant specific list of postulated initiating events (PIEs), possibly combined with additional equipment failures or human errors for specific event sequences definition.	It should be pointed out that the list of PIE should be plant specific.	X			
France 8	3.15 Bullet 1 Bullet 2	- If the initiating event is a failure of part of an electrical distribution system, the AOO, DBA or design extension conditions analysis should assume the unavailability of all the equipment powered from that part of the distribution system - If the initiating event is an energetic event, such as the failure of a pressurized system that leads to the release of hot water or pipe whip, the definition of the AOO, DBA or design extension conditions should consider potential failure of the equipment which could be affected	Failure of part of an electrical distribution system can lead to AOO Same for failure of pressurized system	X			
Switzerlan	3.19	PSA insights and results…	Use of PSA insights and		*Bullet 1:*		*PSA is not available*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
d 2			results should be counted at first since it gives the deepest information of PIE's and their occurrences. Also, the engineering judgement in the establishment process of the design basis is to be mentioned as an important tool.		- Use of analytical methods …, failure modes and effects analysis (FMEA), engineering judgement and master logic diagrams		*at the beginning of a new design, it cannot be used as the major input for the list of PIEs*
Germany 7	3.19	3.19. The set of PIEs should be identified in a systematic way. This should include a structured approach to the identification of the PIEs such as: - Basis for the determination of the plant specific list of PIE should be the event spectrum determined by the vendor of the plant under examination; - Use of analytical methods such as hazard and operability analysis (HAZOP), failure modes and effects analysis (FMEA), and master logic diagrams; - Comparison with the list of PIEs developed for safety analysis of similar plants (ensuring that prior flaws or deficiencies are not propagated); - Analysis of operating experience data for similar plants; - Use of PSA Level 1 and Level 2 insights and results.	As starting point for the development of the plant specific list of PIE the event spectrum of the plant developed by the vendor of the plant which should be available should be used. After that, the list must be modified by using the following mentioned steps.				*Event spectrum determined by the vendor is typically used [should be] when it is available. Nevertheless, recommendations provided apply also to the vendor; it seems better not to identify vendor's list as an input.* *PSA use is out of the scope of this Safety Guide*
Czech 6	3.20	… accidents without careful analysis and assessment of the potential impact	dtto No5 comment			X	*See SSR2/1 Req. 20, §5.27*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		on early ~~or~~ large releases.					
Observer ENISS-19	3.20	Ask for clarification	In some countries, certain limiting faults are excluded from the DBAs on the basis of specific justifications such as break preclusion approach. Req. 3.20 is not crystal clear. What is required? Is it required to demonstrate that the excluded events have a negligible contribution to the risk of large or early releases? The requirement should be written more clearly.		*It is indicated:* "… should not be excluded from this category of accidents without careful analysis…"		
Observer EC/JRC-35	3.20 Line 4	… accidents without careful **quantitative** assessment of the potential contribution to **overall risk figures of merit**.	Same reasons stated in previous comment 32		*Correction:* "…Secondary system pipe break…" *Last part made consistent with wording used in 3.7. See EC/JRC-33:* "… without careful analysis and quantitative assessment of its ~~the~~ potential of contribution to the overall risk, including to conditions arising that could lead to an early radioactive		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					release or a large radioactive release"		
Switzerland 3	3.21	….normal operation should be considered as PIEs…..	As added	X			
Canada 10	3.21	3.21. Failures occurring in the supporting systems that impede the operation of systems necessary for normal operation should be also considered PIEs if such failures ~~eventually require the actuation of the reactor protection systems~~ directly lead to challenging safety functions and eventually to a threat to barriers against radioactive releases.	Most AOOs do not require actuation of the protection system, but they must nevertheless be analysed. Clauses 3.17 covers this already, but if it is necessary to repeat it, please use the same words.		…if such failures require protective actions		
Observer EC/JRC-36	3.21 Line3	the control and limitation systems.	According to the terminology used in 3.3, reactor protection system is included within the control and limitation systems. Unless distinction is wanted to be made here to limit PIEs related to supporting systems only to those leading to scram, same nomenclature should be used here.		*See Canada-10* …if such failures require protective actions		
Observer EC/JRC-37	*New*	Identification of PIEs applying to AOOs, DBAs and DECs should be carried out on a plant-operational-state basis.	Para 3.9 list of generic operational states should apply to the entire 'POSTULATED INITIATING EVENTS' section when talking about PIEs identification;				*See change in 3.22*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			otherwise explicit mention in this respect should be made on PIEs identification dealing with AOOs, DBAs and DECs.				
Germany 8	3.23	3.23. All PIEs should be subdivided into representative groups of event sequences taking into account the expected frequency of occurrence and its effect on the nuclear power plant. This approach allows the selection of the same acceptance criteria and/or initial conditions in each group, applying the same assumptions/methodologies, and identification of the worst accident (bounding case) in each group.	It is not clear what is meant with "representative groups of event sequences". E. g. the German understanding is that for each level of defense an own set of acceptance criteria exists. The suitable set of acceptance criteria will be applied to each event grouped into the level of defense under examination. Does group mean level of defense? If yes, does the last sentence mean that only one bounding case should be analyzed for each level of defense? An adjustments of the expressions groups, categories, plant state (see table under 3.26) etc. used in the document should be adjusted.		*The content of §3.23 and §3.30 will be combined, resulting in the new §3.23 and §3.24 as follows.* *The wording of these two new paras also answers to other comments made about the same subject:* 3.23. ~~All~~ PIEs should be subdivided into representative groups of event sequences taking into account physical evolution of the PIEs. ~~the expected frequency of occurrence and its effect on the nuclear power plant.~~ These groups gather event sequences that lead to a similar threat to the safety functions and barriers and the need for similar mitigating systems to drive the plant to a safe state. Therefore they can be bound by a single representative sequence which is usually referred to when dealing with the		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					group (and often identified by the associated PIE itself). Then these groups are also categorized according to their frequency of occurrence (see § 3.26). This approach allows the selection of the same acceptance criteria and~~/or~~ initial conditions ~~in each group, applying~~ and the application of the same assumptions and methodologies to all PIEs grouped under the same representative event sequence.~~, and identification of the worst accident (bounding case) in each group.~~ ~~3.30. Groups of PIE should be further subdivided according the mechanisms affecting the performance of the safety functions and integrity of the physical barriers. Special groups of sequences can be thus formed~~ 3.24 Representative event sequences can also be grouped by type of sequences with focus on reduced core cooling and RCS pressurization,		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					containment pressurization, radiological consequences, or pressurized thermal shocks. For instance the PIEs "stop of a MFW pump", "stop of all MFW pumps", "isolable break on MFW system" are all typically grouped under a single representative event sequence which is "Loss of Main Feed Water" which belongs to the "Decrease in reactor heat removal" type of sequence.		
Czech 7	3.23	All PIEs should be subdivided into representative groups of event sequences taking into account the expected frequency of occurrence and its effect on the safety of the nuclear power plant.	Text clarification.		*See resolution to Germany-8*		
Observer EC/JRC-38	3.23/2	… and its effect on the nuclear power plant**, i.e. similar mitigating systems needed to drive the plant to a safe state**.	'effect on the nuclear power plant' should be clarified.		*See resolution to Germany-8*		
Germany 9	3.24	3.24. The postulated initiating events associated with anticipated operational occurrences and DBAs should reflect specifics of the design, but typically should belong to the following types of transients: □ Increase or decrease of the heat removal from the RCS; □ Increase or decrease of the RCS	The list of event categories should be expanded as shown.		*Now §3.25* - Leaks ~~inside and~~ outside containment;		□ Increase or decrease of the RCS pressure; *(Already covered by increase/decrease in heat removal and increase/decrease of RCS inventory)*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		flow rate; □ Increase or decrease of the RCS pressure; □ Anomalies in reactivity and power distribution in the reactor core or in the fresh or spent fuel storage; □ Increase or decrease of the reactor coolant inventory; □ Leaks in RCS without/with potential containment by-pass; □ Leaks inside and outside containment; □ Loss of offsite power; □ Reduction or loss of cooling of the fuel in the spent fuel storage pool; □ Release of radioactive material from a subsystem or component (typically from treatment or storage systems for radioactive waste).					□ Leaks in RCS without/with potential containment by-pass; *(Without is a LOCA; previous bullet)* □ Loss of offsite power; *(Covered by other bullets (decrease of RCS flow, decrease of the heat removal)*
Canada 46	3.24	Suggest an additional bullet, Loss of cooling to fuel during on-power refuelling for PHWR	For PHWR, loss of cooling during on-power refuelling should be considered.	X	*Now §3.25*		
Observer EC/JRC-39	3.24,3.27,3.29/All	Identification of PIEs can be made by attending to events related to challenging different critical safety functions. Within each category of events, PIEs are identified according to plant-specific features. Typical examples of category of events challenging safety functions are the followings:	First, examples shown in 3.24 are classified as "*types of transients*". However, 3.24 and entire section 3 talks about PIEs, i.e. initiating events, so the guide should keep referring to events rather than transient, where the latter could also embrace, as indicated in para 3.23 assumptions and		3.24. The postulated initiating events associated with anticipated operational occurrences and DBAs should reflect the specifics of the design. ~~, but typically should belong to the following types of transients:~~ Some typical PIEs and		

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			acceptance criteria, hence mitigating systems needed. Therefore, I would make a clear distinction between PIE and related derived transient. Otherwise, looking at 3.27, what is the difference between the noun syntagm of every bullet located before and after the colons? Left-hand text is the generic event threatening the critical safety function (primary water level, heat removal, subcriticality, primary integrity, etc.) and right-hand text is the PIE itself. In fact, this is implicitly mentioned in para 3.32 line 2 when referring to 'category of events'. A two-column table could also be included instead of current two-item, slightly unclear lists.		resulting event sequences are suggested in para 3.27 for AOO and 3.29 for DBAs, according to the typical type of sequences listed below: 3.27. Typical examples of PIEs leading to event sequences categorised as anticipated operational occurrences could include those given below, sorted by types of sequences. This list is broadly indicative. The actual list will depend on the type of reactor and the actual design: 3.29. Typical examples of PIEs leading to event sequences categorised as DBA~~s~~ could ~~should~~ include those given below, sorted by types of sequences. This list is broadly indicative.		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					The actual list will depend on the type of reactor and actual design:		
Observer ENISS-20	§3.24/3rd bullet	Anomalies in reactivity and power distribution in the reactor core or in the fresh or spent fuel storage unless these are practically eliminated as presented in paragraphs 7.68 to 7.72 of this Safety Guide;	For the Fuel Building, the safety demonstration associated to reactivity anomalies is based on criticality safety principles with a dedicated referential. As such, they follow a different approach from deterministic studies and should be excluded from this guide.			X	*Practical elimination can be claimed but PIE has to be considered*
Observer EC/JRC-40	3.24	*Remove entire para 3.24*	Para 3.24 is nearly redundant with para 3.27 and 3.29. It does not say anything not accounted for in the other two referred paras.			X	
Czech 8	3.25 Line 2	Special attention should be paid to accidents in which the release of radioactive material could bypass the containment because of potentially large consequences even in the case of relatively small releases of radioactive substances from the core.	Specification of what releases are in mind.			X	*Seems unnecessary*
Canada 11	3.25, last sentence	Moreover, ~~such~~ large bypass accidents do not allow much time for taking action to protect the public in the vicinity of the plant.	Small bypass accidents allow plenty of time to protect the public.	X	Clarification		
Switzerland 4	3.26	Table: DBA Limiting Faults DBC-4, PC-4	It should clearly stated that for existing power plants this frequency range was			X	*This consideration applies to other aspects of the SG.*

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			grouped as beyond design basis accidents in the actual SSG-2. Otherwise this will lead to contradictions in the definition for design extension conditions with the new SSG-2. This frequency range then has to be change for existing plants to DBA which are not designed for.				*§ 1.6 states that the Safety Guide is primarily meant for new NPPs*
Hungary 1	3.26	There is no title of the table, maybe it is Table 1.			"Table 2. Example of AOO and DBA categories used in some MSs"		
Hungary 2	3.26	In the Table 1 is shown PIE categories (frequency ranges) for new built plants, it should be mentioned.	For operating plants there is no DBA3, but DBA range 1E-5 < f < 1E-2 has remained.				*Data provided for illustration (Indicated: "Possible AOO…"; "Indicative frequency range"…)* According to Switz-4, *§ 1.6 states that the Safety Guide is primarily meant for new NPPs*
Canada 12	3.26	Add caption: Table 2 AOO frequency range: 1E-2 < f	Table is not numbered. Should be Table 2. The "f" is missing from the frequency range of AOO. Consider reversing the direction of the frequency ranges, e.g.	X (frequency)	*Table: See Hungary-1 above*		

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			1E-2 > f 1E-4 High to low is more common and more intuitive.				
Japan 4	3.26.	DBC-2, DBC-3, DBC-4 , PC-2, PC-3 and PC-4 are not defined. Should be clarified in footnote or somewhere.	Undefined wording.		*Footnote:* Design Basis Condition (DBC) Plant Condition (PC)		
Germany 10	3.26	3.26. Within each type of PIE, the transients should also be subdivided into categories depending on the frequency of the PIE. Possible anticipated operational occurrences and DBA categories are the following: Table The assignment of each PIE to the frequency ranges has to be checked by an appropriate methodology. For events grouped under plant state AOO, an activation of safety systems for injection and/or heat removal is not allowed. Only operational systems and control and limitation systems are allowed to handle the events.	In case that the grouping of events regarding their frequency is used for the classification of plant states (level of defense?), then the frequency of each event has to be checked in order to confirm the assignment of the events. Furthermore, there must be a demand that all events assigned to plant state AOO don't progress in an activation of safety systems for injection and heat removal.		3.26. For each group of PIE, the representative event sequences should also be subdivided into categories depending on the total frequency of the associated PIEs. The assignment of each PIE to the frequency ranges should ~~has to~~ be checked by an appropriate methodology. Possible anticipated operational occurrences and DBA categories are the following: *Note: Section 3 deals with PIE identification and grouping and not with the acceptance criteria and rules analysis. These*		

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					aspects are addressed in Sections 4 and 7		
Pakistan 2	3.26/ page 13	The column of Table; "Alternative names used in some Member States" may be modified to exclude terms like DBCs and PCs which are not further explained in the document.	Specific practices/ terminologies used by particular Member State(s) may not be used or understandable by other Member States. Also, it is not customary to address different MS practices in the safety standards rather these are depicted in a TECDOC.		*Regarding DBC and PC, see Japan-4.*		*The table can be removed or moved to an annex if so wished by MSs*
Ukraine 3	Para 3.26.	The additional clarifications should be added to the table in para 3.26. What are the meaning of "DBC-2, PC-2", etc. If some examples were provided for DBA, it is recommended to add the relevant examples for DEC, as well.	To clarify the information provided.		*Regarding DBC and PC, see Japan-4.*		*This subsection deals with AOO/DBA*
France 9	3.26	3.26 … the sequence of events ~~transients~~ should …	Better than transients		*Covered in Germany-10 above*		
Observer ENISS-21	3.26	Ask for definitions	DBC or PC categories, used in the table of 3.26, are not defined in the document. It's necessary to define these terms.		*See Japan-4.*		
Observer EC/JRC-41	3.26 Table	N/A	Featured categories in Table 2 of former Safety Guide version should be kept though rows readapted according to type of events included within this subsection, i.e. AOOs and DBAs.			X	

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Pakistan 3	3.27 Bullets 4 and 5 Page 13	Typical examples of PIEs for "Reactivity and power distribution anomalies in the fresh or spent fuel storage" are missing.	Examples of PIEs for "Reactivity and power distribution anomalies in the reactor core" are addressed. In a similar way, PIEs for fresh fuel or spent fuel storage may also be mentioned for completeness and invigorating better understanding.		*Now §3.28* *Bullet 4:* - Reactivity and power distribution anomalies in the reactor core: inadvertent *New bullet 5:* - Reactivity anomalies in the fresh or spent fuel storage: dilution in spent fuel pool *Bullet 7 (now 9):* - ~~Failures of systems ensuring normal operation of fuel pools:~~ Reduction or loss of fuel cooling in the SFP: loss of off-site power… *Bullet 8 (now 10):* - Release of radioactive material ~~from~~ due to leak in RCS with potential containment bypass or from a subsystem or component: minor…		
Japan 5	3.27, 1st bullet	—Increase in reactor heat removal: inadvertent opening of steam relief valves; ~~secondary~~ pressure control	Generalization to include BWR plant.	X			

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		malfunctions leading to an increase in steam flow rate.					
Japan 6	3.27, 3rd bullet	—Decrease in RCS flow rate: trip of ~~one main coolant pump~~ one or more coolant pump(s); inadvertent isolation of one main coolant system loop (if applicable).	Generalization to include BWR plant.	X	*Bullet 3:* —Decrease in RCS flow rate: trip of one or more coolant pumps; inadvertent isolation…(if applicable); start of a main coolant pump		
Canada 47	3.27	Suggest additional example of PIE for PHWR Loss of moderator circulation or decrease or loss of moderator heat sink for a PHWR	For PHWR, moderator system malfunction is an important AOO.	X	*New bullet 6*		
Observer ENISS-22	3.28 - line 2	The subset of PIEs leading to DBAs should be identified. ~~All PIEs identified as initiators of anticipated operational occurrences should also be considered as potential initiators for DBAs.~~ Although … specific reactor.	PIEs identified as initiators of AOOs cannot be DBAs. AOO PIEs consist in frequent events associated to the failure of normal operating functions (as shown with examples given in 3.27) whereas DBA PIEs consist in less frequent events associated with pipe breaks (as shown with examples given in 3.29). As written in 7.33, *"an anticipated operational occurrence by itself should not generate a DBA"*.		"The subset of PIEs leading to DBAs should be identified. All PIEs identified as initiators of anticipated operational occurrences should also be analyzed using DBA rules (see SSR2/1 § 5.75(e)). Although … specific reactor."		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Observer ENISS-23	3.28 and 3.39	*Ask for clarification: inconsistency between 2 paras*	On one hand, 3.28 require to consider as DBA very low frequency events down to a frequency consistent with safety targets and on the other hand, 3.39 require to consider these events as DEC w/o core melt. Clarification is needed.		*§3.28 specifically deals with PIEs for AOO and DBA and §3.39 with those for DEC without significant fuel degradation.*		
Germany 11	3.29	3.29. Typical examples of PIEs leading to DBAs should include those given below. This list is broadly indicative. The actual list will depend on the type of reactor and actual design: —Increase in reactor heat removal: steam line breaks. —Decrease in reactor heat removal: feedwater line breaks. —Decrease in RCS flow rate: main coolant pump seizure or shaft break. —Reactivity and power distribution anomalies: uncontrolled control rod withdrawal; control rod ejection; boron dilution due to the startup of an inactive loop, main steam line break (for a PWR). —Increase in reactor coolant inventory: inadvertent operation of emergency core cooling. —Decrease in reactor coolant inventory: a spectrum of possible LOCAs; inadvertent opening of the primary system relief valves; leaks of primary coolant into the secondary	Completion of the list		*For consistency with last bullet from 3.27 (now 3.28) based on PAK-3, last bullet of 3.29 (now 3.30) will be:* - Release of radioactive material due to ~~from~~ leak in RCS, with potential containment bypass, or from a subsystem or component	X	"main steam line break" *is already listed in the bullet about* "Increase in reactor heat removal "Long lasting LOOP" *is a PIE whereas the list contains types of sequences (different kinds of disturbance of main plant parameters). Impact of LOOP on the plant is covered by the existing list* ("Decrease in reactor heat removal"; "Decrease in RCS flow rate")

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		system. —Long lasting Loss of Offsite Power (LOOP) —Sudden loss of heat removal from irradiated fuel in the fuel pools: a break of piping connected to the water in the pool. —Release of radioactive material from a subsystem or component: overheating of or damage to used fuel in transit or storage; break in a gaseous or liquid waste treatment system.					
Japan 7	3.29, 3rd bullet	—Decrease in RCS flow rate: main coolant pump seizure or shaft break; all coolant pumps trip (for a BWR).	Add items (including BWR)	X			
Japan 8	3.29. 4th bullet	—Reactivity and power distribution anomalies: uncontrolled control rod withdrawal; control rod ejection (for a PWR); rod drop accident (for a BWR); boron dilution due to the startup of an inactive loop (for a PWR).	Add items	X			
Canada 48	3.29	Suggest additional example of PIE for PHWR: Loss of cooling to fuel during on-power refueling for PHWR Loss of moderator circulation or decrease or loss of moderator heat sink for a PHWR	For PHWR, moderator system malfunction and loss of cooling to fuel during on-power refueling are important and unique DBAs.	X			
Observer ENISS-24	§3.29	Typical examples … and actual design: - Increase … - Decrease … - …	"Loss of heat removal" would not only include the loss of the heat removal system, but also the	X	Reduction or loss of fuel cooling ~~of the fuel~~ in the SFP ~~Sudden loss of heat removal~~		"Possibly leading…" *seems unnecessary*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		- ~~Sudden loss of heat removal from irradiated fuel in the fuel pools: a break of piping connected to the water in the pool.~~ Decrease in the pool coolant inventory through a break of piping connected to the water in the pool, possibly leading to malfunctions in decay heat removal systems.	covering offered by the coolant that is ensuring passively the heat removal, which is a scenario to be excluded in a DBA context. The word "sudden" is also not well suited as in the case of a LOOP or of a malfunction in decay heat removal systems studied as an AOO, the loss of forced cooling is also "sudden".		~~from irradiated fuel in the fuel pools: a break of piping connected to the water in the pool~~ Decrease of ~~in the pool~~ coolant inventory due to the ~~through a~~ break of piping connected to the water of ~~in~~ the pool,		
Observer EC/JRC-42	3.31/2	… to their frequency of occurrence and required mitigating systems to drive the plant to a safe state.	PIEs categorization should be consistent with PRA's not only in terms of similar initiating event frequency but also similar event tree family of sequences.			X	*The sentence seems not very clear. A PIE is not a sequence it is just an initiating event, systems necessary to mitigate it are not considered for the categorization*
Observer EC/JRC-43	3.32 Line 4 (addition)	In order to identify the bounding case within a category of events, not only extreme cases should be picked up, e.g. maximum break size; minimum flowrate, but also points placed somewhere in the middle between minimum and maximum values characterizing the spectrum of events within each category.	Sometimes the bounding case is not located at the upper / lower bound of the event group range but somewhere in the middle so that different effects worsening accident evolution are more severe. This might be the case for different SBLOCA evolutions, for instance, with HPIS failure.			X	*This aspect seems covered already. It is stated that "The safety analysis should confirm that the grouping and bounding of initiating events is acceptable." This implies that the bounding case selection should be justified*
Canada 13	3.32, last	Note that a bounding scenario may	Wording is misleading as it	X	*Clarification*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
	but one sentence	combine or amplify the consequences of several PIEs in order to encompass all the possible PIEs ~~grouped together~~ in the group.	could imply that all the PIEs in the group are assumed to happen together.				
Germany 12	3.32	3.32. A reasonable number of limiting cases, which are referred to as bounding or enveloping scenarios, should be selected from each category of events. …	The understanding of the term 'category' is not clear and therefore an adjustment of the expressions is needed (see comment 8).		*Accident categories are defined in §3.26 :* "… or enveloping scenarios, should be selected from each category of events (see .§3.26). These bounding…"		
Observer ENISS-25	§3.34	Handling accidents with irradiated fuel and ~~spent~~ fresh fuel should also be evaluated.	Seems to be redundant. Moreover, in the case of MOX fuel, radiological consequences associated to new fuel could also need to be assessed		"3.34. Handling accidents with both fresh and irradiated fuel ~~and spent fuel~~ should also be evaluated…"		
Germany 13	3.35 Bullet (a)	3.35. In addition, there are a number of other different types of PIEs that would result in a release of radioactive material outside the containment and whose source term should be evaluated. Such accidents include: (a) A reduction in or loss of cooling of the fuel in the spent fuel pool (if pool is located outside containment); (b) …	Here a clarification for special types of reactors is necessary, as there are also reactors in operation with spent fuel pools located inside the containment.	X			
Czech 9	3.35 Bullet (c)	An accidental discharge from any of the other auxiliary systems that carry solid, liquid or gaseous radioactive	For example fire of bitumen product during radioactive waste solidification process	X			

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		material;	or storing.				
Observer ENISS-26	§3.35	In addition, there are … include: (a) A reduction in or loss of cooling of the fuel in the spent fuel pool (if leading to boiling); (b) Reactivity anomalies in the fresh or spent fuel unless practically eliminated as presented in paragraphs 7.68 to 7.72 of this Safety Guide;	(a) In most cases, a partial loss of cooling in the spent fuel pool does not lead to boiling and as such, does not lead to any radiological release. (b) Is the word "storage" missing? In any case, a reactivity anomaly in the fresh or spent fuel storages leading to the release of radioactive material would correspond to a criticality accident, that has to be excluded, and for which the source term is difficult to assess.			X	*See ENISS-20 to 3.24.* *Practical elimination can be claimed but PIE has to be considered*
Germany 14	3.36	3.36. The frequency associated to a type of anticipated operational occurrences or DBA should combine the frequencies of all PIEs that have been grouped together.	The link to frequencies is not clear. Furthermore, the relevance of determination of frequencies of PIEs in the frame of deterministic event analyses is not clear. An adjustment of the expressions is needed (see comment 8) because it is unclear what "grouped together" means.	X	3.36. The frequency associated with a bounding event sequence belonging to ~~to a type of~~ AOO or DBA should use the bounding frequency established for the ~~combine the frequencies of all~~ PIEs that have been grouped together. *Note: 3.26 could be merged also with the new 3.23-3.24 (which*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					replace the former 3.23 and 3.30)		
Canada 14	3.36	3.36. The frequency associated to a type of anticipated operational occurrences or DBA should ~~combine~~ bound the frequencies of all PIEs that have been grouped together.	To "combine the frequencies" suggests adding them. I think "bound" was intended.		*See answer to German y-14*		
Observer EC/JRC-44	3.36/2 (addition)	… according to a similar plant evolution and / or safety systems needed to drive the plant to a safe state.	For clarification's sake.			X	*See resolution to Germany-8, about §3.23*
Observer WNA 1	3.37	with the objective to prove, on the one hand, that core melt can be prevented for any accident sequence that has a significant probability of occurrence and, on the other hand, that the consequences of postulated core melt can be limited. For this purpose, specific design provisions can be defined with the aim either to prevent or to mitigate these sequences.	The aim of DEC-A is not to design specific provisions, it is to prove that there is no shortage in the deterministic analysis			X	*Covered by the reference made to SSR2/1 (Rev.1) Req. 20*
Canada 15	3.38	3.38. Two separate categories of design extension conditions ~~should~~ may be identified~~, using different acceptance criteria and different rules for deterministic safety analysis~~: design extension conditions without significant fuel degradation and design extension conditions progressing into core melt, i.e. severe accidents. Different acceptance criteria and different rules for deterministic safety analysis may be used for these	SSR-2/1 does not require that two categories are created – this is more of an analytical convenience. In particular, SSR-2/1 does not require different rules and acceptance criteria for DEC-A and DEC-B.		3.38. Two separate categories of design extension conditions should be identified~~, using different acceptance criteria and different rules for deterministic safety analysis~~: design extension conditions without significant fuel degradation and		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		categories.			design extension conditions progressing into core melt, i.e. severe accidents. Different acceptance criteria and different rules for deterministic safety analysis may be used for these categories.		
Observer EC/JRC-45	3.39/2	… should take into account those low-frequency, challenged-safety sequences not meeting with DBA postulated conditions, e.g. single-event failure yet ultimately preventing core damage. For this purpose, Level 1 PRA constitutes the most suitable tool due to the comprehensive nature of the delineated accident sequences where no deterministic hypothesis on PIE and subsequent accident evolution has been made.	A structured approach for DEC identification is highly recommended to avoid the unmanageable situation of tackling with hundreds of scenarios when multiple failures are considered. Moreover, related frequencies in multiple failure events are not easily obtained so that –again– PRA becomes twice useful for DEC PIE identification.			X	*The list of DEC-A to be considered is provided in 3.40. PSA is a useful tool for existing plants; regarding new plants it is not available at the time where DEC features have to be developed*
France 10	3.40 Bullet 2, line 3	"…Without actuation of the ~~high~~ pressure safety injection…."	Not restricted to 'high' (eg 'middle')	X	***(??) <u>HP</u>****SI is "a typical example" for some designs, nevertheless it can be removed*		
Observer ENISS-27	3.40	A deterministic …should include: • Initiating events that could lead to situations beyond the capability of the safety systems that are designed for ~~a single initiating event. A typical~~	One major point missing here is that the Design Extension Conditions to consider should be credible enough, with respect to the probabilistic safety targets.		…should include: • Initiating events that could lead to situations beyond the capability of		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		~~exemple is the multiple tube rupture in a steam generator of PWR.~~ DBAs, and whose estimated occurrence frequency is credible enough with respect to probabilistic safety targets, • Frequent AOOs or DBAs (…) • Credible Multiple Failure PIEs (…)	There is no sense in studying single initiating events or sequences that have negligible contributions to the core damage frequency. In addition, in plants where there are safety-related systems specifically designed for the handling of certain DEC events, the proposed wording would reclassify these DEC events outside the scope of DEC analysis, which is not the purpose of this paragraph.		the safety systems that are designed for DBAs. A typical example… • AOOs or frequent DBAs (…) • Credible Multiple Failure PIEs (…)		
Pakistan 4	3.41 Last bullet, Page 17	Multiple failure PIEs are given in generic form. These may be more specific with respect to failure of associated components and mitigating systems.	To better understand the process of modeling and analysis.			X	*Multiple failure sequences are defined at a function level, it is difficult to be more specific as it depends on the plant model.*
Japan 9	3.41, 2nd bullet, 1st item	- anticipated transient without scram (ATWS): anticipated operational occurrences combined with the failure of rods to drop or to insert (does not apply to PHWRs)	Generalization to include BWR plant.	X			
Belgium 2	3.41 Bullet 1, Item 1	Give another example for a "very low frequency initiating event"?	We are not convinced that uncontrolled level drop at midloop is a "very low frequency initiating event".		*We agree. That example is deleted*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Germany 15	3.41 Bullet 1, Item 1	3.41. Although design extension conditions are, to a large extent, technology and design dependent, the list below should be used as preliminary reference of design extension conditions without significant fuel degradation and to be adapted plant specifically: very low frequency initiating events typically not considered as DBA - uncontrolled level drop during mid-loop operation (PWR) or during refuelling	From the experience - at least with German design PWRs - the occurrence of the level drop during mid-loop operation is not an event with a very low frequency. It has got also a relevant contribution in the Level 2 PSA and for German PWRs the event is treated as a DBA. The classification as an event with very low frequency should be checked again.		*We agree. That example is deleted*		
Czech-10	3.41	uncontrolled level drop during mid-loop operation (PWR) or during refuelling	Explanation of term mid-loop operation below the line is recommended.	X	*See Belgium-2 and Germany-15. That example is deleted*		
Czech 11	3.41	total loss of ~~normal~~ fuel pool normal cooling and potential subsequent loss of inventory	I feel differences between wording normal fuel pool cooling versus fuel pool normal cooling.		"total loss of normal cooling in the fuel pool ~~cooling~~ and potential…"		
Russia 1	3.45/ Page 17	3.45. A selection of specific sequences with **fuel** melting (severe accidents) should be made in order to establish the design basis for the safety features for mitigating **fuel** melt accidents	Severe accidents are possible generally speaking outside of reactor core – e.g. in spent fuel pool.		*"core melting" is the term used in [SSR-2/1 (Rev1), Definitions, page 65]. §3.45 will be modified as follows:* 3.45. A selection of specific sequences with core melting (severe accidents) should be made in order to establish the design basis for the safety features for		

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					mitigating core melting accidents, according to the plant safety objectives		
Observer EC/JRC-46	3.45	*Same comment than 45* (just by replacing Level 1 by Level 2).	Same rationale than 45 applies here but applied to DEC with core melting.			X	*DEC-B is deterministic and corresponding conditions are postulated regardless the estimated frequency. The major physical phenomenon have to be addressed*
Germany 16	3.46	3.46. Deterministic safety analysis should consider that the features to prevent core melting fail or are insufficient and an accident sequence will further evolve into a severe accident. Some representative sequences should be selected by adding additional failures or incorrect operator responses to the DBA or design extension conditions sequences, ~~and to~~ by using the dominant accident sequences identified in the Level 2 PSA and by selecting scenarios of Level 2 PSA with large releases independently from their frequencies.	For selection of possible sequences also scenarios of Level 2 PSA with large releases should be considered independently from a very low frequency.			X	*DEC-B is deterministic and corresponding conditions are postulated regardless the estimated frequency. The major physical phenomenon have to be addressed* *Additionally, PSA is not available at the beginning of the design, when severe accident conditions have to be defined. Selection of scenario independently of frequency is mentioned in § 3.49*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
USA 5	3.47 (Pg. 18) Line 2	~~Out of the~~ The representative sequences with core melt (design extension conditions with core melting) should be analyzed to determine limiting conditions, particularly those that could challenge containment integrity, and these conditions should be used ~~the enveloping one should be postulated~~ to provide input to the design of the containment…	Different sequences will provide different limiting conditions. For example, hydrogen combustion provides a different challenge to containment than core melt ejection and direct containment heating.		"~~Out of the~~ ~~r~~Representative sequences with core melt (design extension conditions with core melting), regarding each criteria, should be analyzed to determine limiting conditions. Particularly, those that could challenge containment integrity which should be used ~~the enveloping one should be postulated~~ to provide input to the design of the containment and…"		
Observer EC/JRC-47	3.47/1	*Remove first sentence*	First sentence is already included in first sentence of para 3.48.	X			
Observer EC/JRC-48	3.47 Line 2	*Replace current sentence by the following text:* Core melting scenarios result from safety systems failing to succeed in performing their intended safety function. DBA scenarios, alongside DEC without significant fuel degradation, in combination with mitigating system failures and leading to extended core damage, constitute a long list of scenarios highly difficult to	According to the suggested text, not only one bounding sequence but more than one exists in the field of severe accidents. In fact, recent applications facing such severe-accident identification process have made use of Level 2 PRA one way or another. This is a very sound		*See resolution to USA-5 above. And Germany-16 (this regarding the availability of PSA)* *Suggested wording may be considered too complex but the idea of defining representative*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		handle with. Moreover and contrary to DBAs, bounding sequences will be different depending on the severe accident acceptance criteria. DBA standard technical criteria, such as maximum PCT or clad oxidation, constitute a set of intimately related variables so that conditions leading to one variable maximization will likely lead to other variables maximization. However, this is not the case for severe accidents where related acceptance criteria can be constituted by highly independent variables to an extent that maximization conditions for one surrogate variable means minimization conditions for another. One typical example could be containment hydrogen concentration whose maximization will hardly be bounded by containment pressure bounding sequences. Therefore, a structured approach should be employed here for severe accidents identification. One very useful tool may come from Level 2 PRA so-called Plant Damage States, which constitute a comprehensive set embracing the entire spectrum of severe-accident phenomena embedded in risk-significant (looking backwards) groups of sequences leading to core damage and (looking forwards) featuring similar evolutions in containment.	comment with important consequences so please treat it carefully, not paying unnecessary attention to details related to the suggested format of the para.		*sequences for each criterion is kept.*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Canada 16	3.48, bullet 1	Loss of core cooling capability, such as an extended loss of off-site power with partial or total loss of on-site AC power sources (exact sequence is design dependent), or/and the loss of the ~~main~~ ultimate heat sink	Use of "main ultimate heat sink" implies that there is a secondary UHS. If that is the case, there would be no core melt.		"…or/and the loss of the ~~main~~ normal access to the ultimate heat sink"		
USA 4	3.49 (p. 18)	Replace 3.49 with: The low probability of the failure of successive barriers designed to contain the source term from release to the environment should not preclude consideration of an early or a large radioactive release. Deterministic safety analyses should demonstrate that, as the successive barriers are assumed to fail, the design and response of the nuclear power plant and operators can reasonably be shown to prevent (practically eliminate) accidents that would breach the last barrier to an early radioactive release or a radioactive release large enough to require long-term protective measures and actions.	Care should be taken to assure that the guidance in the standard does not stifle innovation that could lead to safer plant designs. The existing text implies that, even if one could design a reactor in which core melting is not expected to occur, one would still have to have structures, systems, and components that would contain a melting core. This demonstrates that this draft guide is not technology neutral, but is a water-cooled reactor based standard.			X	*1) The main framework of this SG is defined by SSR-2/1 (Rev.1) and Req.20,* §5.30 applies. *2) According to §1.6 of this SG, it "*focuses primarily (…) design safety of new NPPs (…). The guidance provided is (…) it is particularly based on experience with DSA for water cooled reactors. *3) Graded approach is applicable.* *5) Innovative designs may be taken into account in further revisions of the Safety Requirements and consequently in the ones of this SG.*
Ukraine 4	Para 3.50. Line 1	The statement "Severe accident sequences should be selected to identify the most severe plant parameters resulting from the severe	According to the para, the parameters caused by the severe accidents are to be considered in the design of		"3.50. Severe accident sequences should be selected to identify the most severe plant		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		accident phenomena for to be considered in the design of the plant structures, systems, and components t***hat are necessary for preventing such conditions from arising, or, if they do arise, for controlling them and mitigating their consequences***".	all SSC. This statement is too strong, and should be applied for those SSC which are needed for severe accident management		parameters resulting from the severe accident phenomena to be considered in the design of the plant structures, systems, and components that are necessary to limit the radiological consequences of such severe accident sequences."		
Observer EC/JRC-49	3.50/3 (addition)	Special attention in identifying severe accident scenarios should also be paid in the frame of equipment qualification through survivability analysis in order to suitably pick the bounding environmental profiles of the figures of merit which typically are temperature, pressure, humidity, flammable gas concentration and radioactivity.	Environmental qualification under harsh conditions such as those typical of severe accidents should be mentioned here since this is a crucial issue deserving special treatment where ongoing international efforts are under development.		*It will be added:* "…The environmental conditions should be taken into account in the qualification of equipment used in severe accidents. "		
Japan 10	3.51. /L3 and others	Analysis of internal and external hazards differs from analysis of postulated initiating events and scenarios originated by a single failure or multiple failures in the nuclear power plant technological systems or by erroneous human actions having direct impact on performance of ~~fundamental~~ main safety functions.	In accordance with the IAEA Safety Glossary, the functions formerly named 'fundamental safety functions' are now named 'main safety functions'.		*A foot note will be added:* (*) According to the IAEA Safety Glossary (2016) the term "main safety functions" is equivalent		
Belgium 1	1.8 and **3.51**	Make article 1.8 and articles 3.51 till 3.54 coherent.	At one hand, art. 1.8 says that internal and external		*Note: See changes to §1.8 in Section 1.*		

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			hazards are not covered. At the other hand, article 3.51 till 3.54 cover these hazards. This seems not coherent.		"3.51. Determination of PIEs should consider effects and loads from events caused by relevant site specific internal and external hazards …". *Note: It is the purpose of §3.51 to clarify that hazards are not PIEs by themselves but their effects and loads can induce PIEs and the analysis of these PIEs should take due account of their origin.*		
Observer ENISS-28	3.51/3	Ask for clarification : Determination of PIEs should …. A list of examples external hazards can be found in NS-R-3 [14]	Reference [14] is under full revision (step 5 in April 2016), and contents of the modifications are not known. Is that clear that [14] refers only to the current published version?		*This reference will be updated at the moment to publish the SG (SSG-2) according to the publication available in that moment (not drafts are referenced in published SGs).*		
Japan 11	3.52.	Please consider making 3.52 more specific guidance. One idea is to add examples that should be taken into account such as loss of electrical grid, loss of ultimate heat sink,	There are no specific guide for safety analysis of multiple uniti plant sites. Just only repeats SSR-2/1 (Rev. 1).		*It will be added:* "…into account. Specifically, the effects from losing the electrical grid, those from losing the		

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		failure of shared equipment.			ultimate heat sink and the failure of shared equipment should be taken into account.”		
Switzerland 5	3.53	- such hazard can be screened out due to its negligible contribution to risk, or	A definition of “negligible contribution to risk” should be added or referenced.		“3.53 The analysis of hazards which is performed by using probabilistic methods or appropriate engineering methods [a Reference will be added] should demonstrate…”		*Definition of "negligible contribution to risk" is out of the scope of this SG but has to be assessed in [hazard] dedicated guides*
Czech 12	3.55	Event sequences that lead to early or large radioactive releases[5] are required to be practically eliminated	Use this (5) below the line explanation in para 2.1 where wording early or large is used for the first time, if my comment against using his wording “early or large” will not be accepted..		*See other resolutions, e.g. 2.1 (CZ-2 and other) and 2.18.* “…minimized. Conditions arising that could lead to an early radioactive release or a large radioactive release”…		
Observer EC/JRC-50	*After* 3.55/New	According to SSR-2/1, Rev. 1, two types of source term release scenarios should be 'practically eliminated': large release and early release category. Since severe accident consequences on source term magnitude, composition and timing to determine whether a particular scenario should be classified under one of the two abovementioned categories is a very complex issue,	Para 3.56 should be deeply improved: First, classification attending to 'events' and 'severe accident phenomena' does not fit well with identifying conditions leading to large or early release. Second, it is not mentioned how plant-specific this issue			X	*The clarification seems not necessary and out of the scope of this SG. On the other hand it is severe accident oriented; some DBA are also excluded because of the practical elimination*

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		exhaustive identification of scenarios belonging to both categories can be made through Level 2 PRA so-called Release Category Figure of Merit whenever available.	is, but this should be remarked. Third, 1.a event, i.e. 'failure of large pressure-retaining component in the RCS' is not a very common methodology; it this is referring to LBLOCA, containment related failure will most likely occur because of containment overpressurization, which seemingly falls under the late containment failure category 3; however, containment failure times in LBLOCA w/o any safety systems can lead to very early releases; 2.c on hydrogen DDT can also happen in the long ex-vessel phase by building up of flammable gases thereby falling again under point 3 of the classification. As a conclusion, I would remove entire para 3.56 or rewrite it completely (please look at suggested text in following comment 52)				*objective*.
Observer EC/JRC-51	3.56/All	Conditions leading to early and large releases highly depend on plant-specific features, e.g. mitigating systems performance, containment characterization, etc., and regulatory	See previous comment rationale			X	*See EC/JRC-50*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		as-defined categories of what is meant by 'early' and 'large' release. Notwithstanding the above, several scenarios in particular present significant contributions to both categories whose elimination will hence help achieve the 'practically eliminated' objective: 1) Early releases: a. Uncontrolled reactivity transients; b. High-pressure RPV failure (potentially leading to Direct Containment Heating hence jeopardizing containment mechanical integrity); c. Containment isolation failure; d. Containment bypass: Interfacing System LOCA (ISLOCA), both as initiating event and at recirculation switch; SGTR e. Steam Explosions: In-Vessel explosions (so-called ALPHA mode) whose latest state of the art has estimated this phenomenon to be 'practically eliminated'; and Ex-Vessel at RPV failure in case of wet pedestal / reactor cavity configuration. However, steam explosions go beyond the operator control, i.e. no mitigating human action or equipment can be implemented to avoid such severe-accident phenomena. 2) Large releases: a. Aside from the scenarios mentioned above, all kinds of containment failure					

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		may lead to severe source term releases in the long term as a consequence of losing the last defence-in-depth barrier.					
Germany 17	3.56, Line 1, page 19	3.56. The event sequences requiring specific demonstration of their "practical elimination" should be classified as follows, ~~if need be with a design specific adaption~~:	Is the colored part of the sentence necessary? Deletion improves readability.	X			
Japan 12	3.56 (group 2)	2) Severe accident phenomena which could lead to early containment failure: a. Direct containment heating b. Large steam explosion c. ~~Hydrogen detonation~~ Large hydrogen explosion	It is not ensured solely detonation will lead to containment failure.		"c. Explosion of combustible gases, including hydrogen and carbon monoxide"		
Japan 13	3.56 (group 3)	3) Severe accident phenomena which could lead to late containment failure: a. Molten core concrete interaction (MCCI) b. Loss of containment heat removal c. Large hydrogen explosion	Hydrogen explosion is not limited in early phase.	Partially	"c. Explosion of combustible gases, including hydrogen and carbon monoxide"		
Germany 18	3.56, Grop 3 Page 20	3) Severe accident phenomena which could lead to late containment failure: ~~a. Molten core concrete interaction (MCCI)~~ b. Loss of containment heat removal	Current experiences with severe accident analyses for different reactor types have shown that late failure of containment by MCCI cannot be practically eliminated, especially for older plant designs. May be for next generation plants		*See France-11*		*It should be practically eliminated in new design, otherwise radiological consequences of SA cannot be limited*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			like EPR that might be possible. Checking if that can be listed here as a scenario which can be practically eliminated.				
France-11	3.56 2)	2) Severe accident ~~phenomena which~~ events that could lead to early containment failure: a. Highly energetic direct containment heating b. Large steam explosion c. Hydrogen detonation or deflagration with impacts exceeding the containment capacity 3) Severe accident conditions ~~phenomena~~ which could lead to late containment failure FNXX: a. Basemat penetration or containment bypass during molten core concrete interaction (MCCI) b. Long term loss of containment heat removal leading to an uncontrolled failure of the containment 4) Severe accident with containment bypass 5) Significant fuel degradation in a storage pool and uncontrolled release FNXX – These conditions should be analysed during the identification of situations to practically eliminate. Nevertheless, it should be generally practicable to mitigate them.	It should be better to consider in the safety analysis severe accidents which could lead to late containment failure and to mitigate them according to DiD because for most of them,in particular for new reactors, mitigation is possible The text above is not related to phenomena		*See Japan 12 and 13* 2) Severe accident sequences that ~~phenomena which~~ could lead to early containment failure: a. Highly energetic ~~D~~direct containment heating b. Large steam explosion c. ~~H2 detonation~~ Explosion of combustible gases, including hydrogen and carbon monoxide 3) Severe accident sequences that ~~phenomena which~~ could lead to late containment failure: a. Basemat penetration or containment bypass during ~~M~~molten core concrete interaction (MCCI) b. Long term of ~~L~~loss		- *"events" in 2 and "conditions" in 3 harmonized (events is used in the SG).* - *It seems better to use "explosion" only in this SG and not also deflagration and detonation.* - *2 (c) "impacts exceeding the containment capacity" is included in the title. Similar for 3 (b)* - *Foot note seem out of the scope of 3.56*

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					of containment heat removal (…) 4) Severe accident with containment bypass 5) Significant fuel degradation in a storage <u>fuel</u> pool <u>and uncontrolled releases</u>		
Observer WNA 2	3.57	3.57. Consequences of event sequences that have been 'practically eliminated' do not need themselves to be deterministically analysed. ~~Nevertheless, severe accident management guidance for "not postulated scenario' should be provided,~~ but their 'practical elimination' should be demonstrated, including relevant deterministic analysis, as presented in paragraphs 7.68 to 7.72 of this Safety Guide.	No guidance can be provided for events that are not analyzed		*The sentence will be reformulated or deleted*		*Bottom line:* "Consequences of accidental conditions <u>that lead to early/large releases</u> (i.e to be pr. el.) do not need themselves to be deterministically analysed, but their practical elimination should be demonstrated (7.68 to 7.72)
Canada 49	3.57	Suggest a definition be provided for "not postulated scenario"	A definition for "not postulated scenario" is not available in this document		*See WNA-2.* *The sentence will be reformulated or deleted*		
Observer EC/JRC-52	3.57/2	N/A	Second sentence should be rephrased or removed. Its current meaning is unclear: what does 'not postulated scenario' mean, even more when talking about a		*See WNA-2.* *The sentence will be reformulated or deleted*		

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			dedicated SAMG aimed at such scenario? Paras 7.68-7.72 describes pertinent suggestions to conduct deterministic analysis for ' practically eliminated' scenarios. But these rules should not ever been referred as 'severe accident management guidance' for obvious reasons.				

Section 4
DS491 Draft Safety Guide: Deterministic SA for NPPs - Step 7

COMMENTS BY REVIEWER				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepte d	Accepted, but modified as follows	Rejected	Reason for modification/rejecti on
France 12	4.2 Line 3	"… Compliance with the ~~deterministic~~ acceptance criteria…"	Delete 'deterministic' in acceptance criteria	YES			
HUN 3	4.3	4.3. *Acceptance criteria should be established for the entire range of operational states and accident conditions~~, including severe accidents~~. These criteria should aim at limiting damage to barriers against the release of radioactive material in order to prevent unacceptable radiological releases. Selection of the criteria should ensure sufficient margin between the criterion and the physical limit for loss of integrity of a barrier against release of radioactive material.*	This is really true for DBA cases, but last sentence may not be fulfilled for DEC2 severe accident so phrase "including severe accidents" should be deleted.	YES			
CAN 50	4.5 Bullet 2	Suggest the following changes, *Detailed/derived technical criteria which relate to integrity of barriers (fuel matrix, fuel cladding, RCS pressure boundary, containment) against radioactive releases or technical criteria which can be applied to preclude failure of barriers, e.g. adequacy of coolant*	Other technical criteria may be developed which are not directly related to barrier integrity but represent sufficient but not necessary conditions for the integrity of the barrier.			X	Confusing. It could be a derived criteria for integrity of barriers

		inventory in secondary circuit for PHWR.					
GER 19	4.5	—Detailed/derived technical criteria which relate to integrity of barriers (fuel matrix, fuel cladding, RCS pressure boundary, containment) against radioactive releases. They are typically proposed by the designer and subsequently approved by the regulatory body for use in the safety demonstration.	Detailed/derived technical criteria (e.g. max. cladding temperatures, max. fraction of cladding oxidation, max. hydrogen concentration, etc.) ion, are often regulatory requirements, too.		Addressed in the comment below (FIN-1)		
FIN 1	4.5 *Bullet 2, 2nd sentence*	Detailed/derived technical criteria which relate to integrity of barriers (fuel matrix, fuel cladding, RCS pressure boundary, containment) against radioactive releases. They are defined by regulatory requirements or ~~They are typically~~ proposed by the designer and subsequently approved by the regulatory body for use in the safety demonstration.	Many such criteria (e.g. peak cladding temperature < 1200 C) are defined by the regulatory requirements	YES			
CZ 13	4.5. *Bullet 1, 2nd sentence*	High level (radiological) criteria which relate to radiological consequences of plant operational states or accident conditions. They are usually expressed in terms of ~~releases~~ activities or doses typically defined by law or by regulatory requirements.	Clarity of the text.	(YES)	"…usually expressed in terms of ~~releases~~ activity levels or doses typically…"		
France 13	4.5 Both bullets	- High level (radiological) criteria which relate to radiological consequences of plant operational states or accident	To be in accordance with existing practices.			X	*See comments above (FIN-1, CZ-13).* ***First bullet:*** *The clarification*

		conditions. They are usually expressed in terms of releases or doses typically defined by law or by regulatory requirements. Such criteria can be quantitative or qualitative (for example: no need for emergency protective measures, limitation of consequences in area and time) - Detailed/derived technical criteria which relate to ~~integrity of barriers~~ safety functions …	More general than 'integrity of barriers' as safety function covers confinement which is related to integrity of barriers. It would be worthwhile not to limit criteria to one safety function				*may be not necessary.* ***Second bullet:*** *Maybe too general.*
FIN 2	4.6 *First sentence*	The radiological acceptance criteria should be expressed in terms of effective doses, equivalent doses or dose rates to nuclear power plant staff, general public or *as appropriate* environment, including non-human biota. The doses are required to be within prescribed limits and as low as reasonably achievable in all plant states, SSR-2/1 (Rev.1), Req. 5 [1].	Clarity Add. "as appropriate" It is not common that dose limits are presented to environment, including non-human biota.		"… power plant staff, the general public or the environment, including non-human biota, as appropriate. The doses are required to be within		
CZ 14	4.7	Radiological acceptance criteria expressed in terms of doses may be conveniently transformed into acceptable release~~s~~d activities of~~or~~ different radioactive isotopes in order to decouple nuclear power plant design features from the characteristics of the environment.	Releases are expressed in activities of individual radionuclides taking into account their different radiological risk.		(see CZ 13) "…transformed into acceptable activity levels ~~releases~~ for different radioactive isotopes		

GER 20	4.7	4.7. Radiological acceptance criteria expressed in terms of doses may be conveniently transformed into acceptable releases for different radioactive isotopes in order to decouple nuclear power plant design features from the characteristics of the environment.	Meaning of the colored part of the sentence is unclear.				
CZ 15	4.9 *2nd sentence*	They should be more restrictive than for DBAs since ~~their~~ frequencies of their appearances are higher.	Clarity of the text.			X	The change seems unnecessary
CZ 16	4.10	The radiological acceptance criteria for DBAs to be established should ensure that very restrictive ~~dose~~ design limits, according to Req. 19 § 5.25 from SSR-2/1 (Rev.1) [1], are met.	There are no any "dose" limits in the referred document Req. 19 § 5.25 from SSR-2/1 (Rev.1) [1.]		*Note: Covered by the resolution provided to the comment below (CAN 17)*		
CAN 17	4.10	4.10. The radiological acceptance criteria for DBAs ~~to be established~~ are typically less restrictive than those for AOOs but should ensure that ~~very restrictive dose limits, according to~~ Req. 19 § 5.25 from SSR-2/1 (Rev.1) [1], ~~are~~ is met.	Use of "very restrictive" is questionable since AOO limits are more restrictive (para 4.9).	YES			
14	4.12	Technical acceptance criteria should be set in terms of the variable or variables that govern the physical processes that challenge the ~~integrity of the barrier~~ safety functions. It is a common engineering practice to make use of surrogate variables to establish an acceptance criterion or combination of criteria that, if not exceeded, will ensure the the	More general than integrity of barriers			X	*In this paragraph, the use of "integrity of the barrier" seems more adequate*

		~~integrity of the barrier~~ safety functions. Examples of surrogate variables are: peak cladding temperature, departure from nucleate boiling ratio or fuel pellet enthalpy rise. When defining these acceptance criteria, a sufficient conservatism should be included to ensure that there are adequate safety margins to the loss of ~~integrity of the barrier~~ the safety functions					
GER 21	4.13 *Bullet 4* *Bullet 7*	□ Criteria related to integrity of nuclear fuel located outside the reactor: adequate subcriticality, adequate water level above the fuel assemblies, and adequate heat removal □ … □ Criteria related to integrity of the containment and limitation of releases to the environment: duration and value of maximum and minimum pressure, maximum pressure differences acting on containment walls, avoiding containment low-pressure, leakages, concentration of flammable/explosive gases, and acceptable working environment for operation of systems.	Addition of some criteria for the sake of completeness		***Bullet 4:*** "…the reactor: adequate subcriticality, adequate water inventory ~~level~~ above the fuel assemblies, and adequate heat removal		***Bullet 7:*** *Unnecessary clarification*
CAN 51	4.13 Bullet 2	Suggest the following changes, *Criteria related to integrity of fuel cladding: minimum departure from nucleate boiling ratio, maximum*	For PHWR, the Departure from Nucleate Boiling (DNB) does not generally lead to		***First change*** *accepted* (…nucleate boiling ratio…)		***Second change:*** *Better not to indicate an specific value*

		cladding temperature, maximum local cladding oxidation. For some designs (i.e., a PHWR), the acceptable minimum departure from nucleate boiling ratio may be one.	significant immediate clad temperature increases. Minimum ratio of DNB				
CAN 52	4.13 Bullet 3	Suggest the following changes, *Criteria related to integrity of the whole reactor core: adequate subcriticality, maximum production of hydrogen from oxidation of claddings, maximum damage of fuel elements in the core, maximum deformation of fuel assemblies (as required for cooling down, insertion of absorbers, and de-assembling), calandria vessel integrity (for PHWR)*	For PHWR, the integrity of the calandria vessel is also important to maintain the geometry of the reactor core.	X			
UKR 5	Para 4.13 *Bullet 7*	*To extend the criteria related to integrity of the containment and limitation of releases to the environment with the* "isolation of the containment, maximum temperature in the containment"	To cover all possible criteria for containment integrity		"…environment for operation of systems, ~~isolation of the containment,~~ maximum temperature in the containment"		~~isolation of the containment~~ *(penetrations) covered with "leakages"*
CZ 17	4.14	For postulated initiating events occurring during shutdown operational regimes or other cases with disabled or degraded integrity of any of the barriers, more restrictive criteria should be preferably used, e.g. avoiding	Not valid for fresh fuel.			X	*It may contain UO_2 fuel partially irradiated and MOX*

		boiling of coolant in open reactor vessel or in the spent fuel pool, or avoiding uncovery of spent fuel assemblies.					
ENISS	4.15 *line 2*	In particular, technical acceptance criteria .. with higher ~~probability~~ frequency of occurrence. For AOO there should be	In the whole document, "frequency" should preferably be used instead of "probability".	X			
ENISS	4.15 *line 5*	For DBA, and for design extension conditions without significant fuel degradation, ~~there should be no (or limited) consequential damage to the RCS, containment integrity should be preserved, and damage of the reactor fuel should be limited~~ barriers to the release of radioactive material from the plant should maintain their integrity to the extent required to meet Req. 4.10 or 4.11. For design extension conditions …	As written, this requirement may be misunderstood. Obviously, damages to the RCS are not prevented when the PIE is a LOCA. Containment integrity is not preserved in DEC events with postulated containment bypass.		"… ~~damage of the reactor fuel should be limited~~ barriers to the release of radioactive material from the plant should maintain their integrity to the extent required (see §4.10 and §4.11).		
ENISS	4.17	Although the assessment … with the ~~probability~~ frequency of the loads they have to bear.	See comment 4.15/2			X	

Section 5
DS491 Step 7: Deterministic Safety Analysis for NPPs

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date:				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Observer ENISS2	General comment	A/ Quality of code development and maintenance : from #5.7 to #5.12, #5.40 B/ Verification and Generic Validation Verification : from #5.13 to #5.18 Validation : #5.4 (to be mixed with #5.23), #5.20, beginning of #5.19, from #5.26 to #5.28, #5.30, #5.34 C/ Uncertainty Quantification : #5.21, #5.29, from #5.31 to #5.33, from #6.21 to #6.29 D/ Code documentation : #5.2, #5.38, #5.36, #5.37, #5.39 E/ Adequate use of the code for safety studies Qualification of the code : code fitted to the study (#5.1, #5.4, end of #5.19, #5.22, #5.24, #5.25), accuracy of the results of interest for the study (#5.3, #6.7, #6.26, #6.29) Compliance with the users' guidelines : #5.6, #5.35 Users' technical and scientific					*No specific suggestions*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		competence : #5.5					
Observer ENISS 3	General comment	Examples of terms needing definition: Verification (#5.13 and #9 don't use the same meaning for verification) Validation Review, inspection and audit (#5.14) Error (#5.29) Robust (#5.2(e))			*"error" (5.29) will be replaced by "uncertainty".* *See ENISS-32* *The term "robust" is used in reports such as [10], meaning in general "without oscillations or non-convergence or results with large differences when only small disturbances are input"*		*The use of these definitions is consistent with other Safety Standards, Safety Reports and with the Safety Glossary. Specific definitions for these terms are outside the scope of this SG.*
Canada53	5.1		This clause suggests a graded approach in software qualification such that the requirements for validation and verification depend on the type of application and purpose of analysis. The concept of graded approach can be extended beyond software qualification to the actual deterministic safety				*No specific proposal*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			analysis as well.				
Madagascar 1	5.1	"Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation to a sufficient degree"	The meaning of the sentence can be different if the word "to a sufficient degree" is not included in the reference, it is better to put the full sentence form GS-R Part 4			X	*Requirement 18 from GSR Part 4 (Rev.1) is "Any calculational methods and computer codes used in the safety analysis shall undergo verification and validation".*
Observer EC-JRC 53	5.2/1	Regarding the selection ~~and use~~ of computer codes…	The use of computer codes is not treated in listed bullets of section 5.2. Rather, para 5.6 specifically addresses this topic.	X			
Observer ENISS 32	5.2 (e)	Ask for clarification	What means "robust"? (to include in a Glossary)		*The term is used in reports such as [10], meaning in general "without oscillations or non-convergence or results with large differences when only small disturbances are input"*		
Canada 54	5.3 bullets (b), (c)	Suggest the following changes, *The assessment of the accuracy of individual codes should include a series*	Estimation of uncertainties associated with numerical approaches and key models are not always			X	*As far as possible to avoid compensatory effect, overall code uncertainties should*

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		of steps, some of which are related and may be considered as a whole:	separate steps and their contributions to overall code uncertainties may not readily established.				*not be performed*
Observer EC-JRC 54	5.3/All	*Remove/Replace*	The goal of para 5.3 is unclear: code uncertainty assessment –if this is what pursued as it can be likely derived from bullets (a) to (d)– only concerns BEPU approach hence guidance concerning this issue should not be included without previously making explicit the specific code approach underlying such guidance.			X	*This para is related to accuracy of the results which is to be verified whatever the approach of the code is.*
Observer EC-JRC 55	5.4/All	*Replace*	Entire para 5.4 should be regrouped under para 5.2. Para 5.4 focuses on code validation through benchmarking activities, thereby in intimate relation of para 5.2 bullets on the minimal capabilities to be met by the code in order to be selected.			X	*Paras 5.2 and 5.4 have different objectives. Para 5.2 is for selection of computer codes. Para 5.4 is for validation of the selected computer codes. No need to combine both paras*
Switzerland 6	5.5	(a) The users have received adequate training and that they appropriately understand the code, (b) The users are sufficiently	Full understanding of a very complex code is difficult to achieve by a user.			X	*Changes may not add relevant value*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		experienced in the use of the code and ~~fully~~ suitably understand its uses and limitations,					
PL 1.	5.5. (d)	The users follow the recommendation for use of the code and especially the ones relative to the application ~~the user are carrying out the analysis~~ for which the analysis are carried out	Clarification	X			
Germany 22	5.6 (c)	(c) The ~~nodalization,~~ selected models and assumptions match the ones chosen for SET and IET used for the qualification of the application	The nodalization of a plant modelling will be different to the nodalization for test sections of single effect tests and integral effect tests. E.g. the core region is subdivided into several rings of thermal hydraulics channels, larger amount of fuel assemblies has to modelled, internals of RPV has to be modelled, different injection and discharge of reactor coolant, etc. Thus, the demand of equal nodalization should be deleted.			X	*The nodalization of a plant modelling will be different to the nodalization for SETs, IETs and NPPs. However the consistency of nodalization is necessary.*
PL 2.	5.6. (c)	(c) The nodalization, selected models and assumptions ~~match~~ are consistent with the ones chosen for SET and IET	Consistency is the better word when You describe two nodalizations		"(c) The nodalization, selected models		

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		used for the qualification of the application			and assumptions ~~match~~ are consistent, to the extent practicable, with the ones chosen for SET …”		
Observer EC-JRC 56	5.5, 5.6/All	Even though substantial progress in the development of more accurate and reliable computer codes has been made, user effects still have a dominant influence on the final results. This is why quality assurance to code users dealing with safety analysis applications should be required. Since performing transient simulations in complex system codes basically consists of fitting certain real processes with theoretical models implemented in the code, the main categories where user effects concentrate can be structured in 'reality' and 'code': a. 'Reality' category comprises: a.1. Plant: The user should have very good knowledge of plant characteristics including SSCs performance in order to prepare a good input deck. For instance, deviations in input and boundary conditions can lead to strong deviations in the outputs; a.2. Physics: The user should have very	Importance of user effects has been remarked by many international activities dealing with code uncertainty assessment, e.g. within CSNI, CNRA, European Nuclear Regulators, etc. This Safety Guide instead puts no emphasis on such delicate topic which takes even more important when talking about severe accident codes such MELCOR or MAAP (due to the higher freedom assumed by the user compared to the frame of DBA-oriented codes). Therefore, it is the opinion of this reviewer that a fundamental gap is currently found when stressing how important is			X	*Detail out of the scope of this Safety Guide. Some suggestions not clear (QA for code user)*

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		good knowledge on phenomena governing accident evolution; b. 'Code' category comprises: b.1 *Software*: The user should be fluent in constructing and understanding modelling aspects to build up an input deck. For instance, nodalization mesh plays an important role in adequately capturing the most important phenomena driving accident evolution; b.2. *Hardware*: Not only knowledge on nuclear reactor neutron and thermalhydraulics is fundamental, but also to be familiar with code calculation structure scheme, i.e. employed set of continuity equations or time step size, and code phenomena models. For instance, the user has to make many choices on selecting the most suitable model for a specific phenomenon. Also state and transport property data, i.e. range of reference points for property tables, could be also defined by the user. This user effect source plays an even more critical role in severe accidents where the number of phenomenon where alternative models are available for user's choice hugely increases in proportion to a much lesser reliable state of the art supporting code	that code users are well trained in the three independent fields pointed out in my suggested writing. In fact, different countries such Finland, USA or The Netherlands have already given a step forward and started working in developing quality assurance programs for code users. Suggested text should therefore constitute a new subsection –just, for instance, as 'VALIDATION OF COMPUTER CODES'–.				

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		modelling validation.					
Japan14	5.13-5.15	(new) 5.13.a. In accordance with GSR Part 4 (Rev. 1), § 4.60 [2] verification of the code should consist of model verification and system code verification. (new) 5.13.b. The model verification should be performed by examining solution characteristics and making comparisons of outputs of the code with reference analytical solutions or outputs of other verified code to assure the fidelity of numerical solutions of the code, e.g., time and space discretization, solution symmetry, and dependencies or robustness on initial conditions and boundary initial conditions, etc. 5.14. The ~~verification of the code~~ system code verification should be performed by means of review, ... 5.15. ~~Verification of the code~~ The system code verification should be performed to review the source coding...	To be consistent with GSR Part 4 para 4.60, divide verification into model verification and system code verification. And add paragraph related to the model verification.		(new) 5.13.a. In accordance with GSR Part 4 (Rev. 1), § 4.60 [2] verification of the code should consist of both model verification and system code verification.		*The scope of the para 5.13b suggested does not relates to verification but to validation.* *5.14 and 5.15 are common to both model verification and system code verification. Suggested changes are not applicable*

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Observer ENISS 33	5.13	Verification is the process of determining that a computational model accurately represents the underlying mathematical model and its solution. Verification of the code should be …	The definition of verification is lacking. The proposed new text is internationally accepted. Both code verification and solution verification must be taken into account. Nothing is said about solution verification. It could be integrated in the glossary.			X	*See Japan-14.* *Better not to add this clarification/ definition, which is not related to DSA*
Canada 55	5.14	Suggest the following changes, *The verification of the code should be performed by an independent verifier, by means of review, inspection and audit. Checklists might be provided for review and inspection. Audits might be performed on selected items to ensure quality.*	Verification of computer code should be performed by an independent verifier			X	*It depends on the specific QA procedure from the code development organization. GSR Part 4 (Rev.1) requires independent verification of safety assessment. (Requirement 21).*
Observer ENISS 34	5.14/1	Need of a glossary	“review, inspection and audit”: the definition of these words must be provided			X	*Used according Safety Standards and Safety Glossary*
Belgium 3	5.16	“ … software platform …”	What is a “software platform”? Is it clear for the readers?		“… or software platform (e.g. operating system) other than that…”		
Observer	5.17/2	Verification of the source code should	One can find standards for		“… conforms to		

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ENISS 35		be performed to demonstrate that it conforms to ~~programming standards and~~ language standards, and that is logic is consistent with the design specification.	code development and maintenance but there are no standards for programming (except internal specific standards within a development team)		accepted programming practices- ~~programming standards and language standards…"~~		
Japan15	5.19	... scope of validation might be relaxed for codes used in severe accident analysis, taking into account the limited relevant experimental data. When validation is limited due to above reason, review of model applicability by experts considering experience and the level of knowledge on the model might be encouraged.	Add recommendation where validation is limited.			X	*Out of scope of this Safety Guide. It can be found in dedicated documentation*
France 15	5.19	5.19. Validation of the computer code should provide confidence in the ability of a code to predict, realistically or conservatively, the values of the safety parameter or parameters of interest. The level of confidence provided by the validation should be appropriate to the type of analysis; ~~scope~~ accuracy of validation might be relaxed for codes used in severe accident analysis, taking into account the limited relevant experimental data ; nevertheless,	It is needed to get a reasonable confidence that provisions for severe accident or DEC are efficient.			X	*"scope" is more appropriate than "accuracy". This sentence says that full validation of a severe accident computer code may not be feasible due to limited experimental data.* *Also, the*

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		validation shall be sufficient for the demonstration of the effectiveness to the design provisions.					*requirements for computer codes are established in GSR Part 4 (Rev.1); no requirements can be added here*
Observer ENISS 36	5.19 and 5.20	Reverse 5.19 and 5.20	5.20 (definition of validation) should come before 5.19 and include the first sentence of 5.19. We suggest to reverse their order in the document.	X			
Observer ENISS 37	5.19/4	Validation of the computer code … type of analysis; scope of validation might be relaxed for codes ~~used in severe accident analysis, taking into account the~~ with limited relevant experimental data (for example, codes used in severe accident analysis).	The recommendation is larger than the scope of severe accident			X	*Validation cannot be 'relaxed' for codes used in DBA*
Canada 56	5.21 Line 2	Suggest the following changes, *Outputs of the code are compared with relevant ~~experimental data~~ measurements from tests or operational transients for important phenomena expected to occur.*	As noted in para 5.23, nuclear power plant transients should also be used in addition to experimental data for separate effect tests and integral effect tests.		**"…**Outputs of the code are compared with relevant experimental data and with operational transients, if possible, for the important phenomena expected to occur."		"experimental data" *is also used in other paragraphs. Better not to change it.*

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Observer ENISS 38	5.21/1	5.21. Validation of the code ~~should be performed~~ may help, when the conservative approach is not sufficient, to assess the uncertainty of values predicted by the code. Outputs of the code are compared with relevant experimental data for important phenomena expected to occur.	The aim of validation is not uncertainty quantification. The acronym VVUQ (Verification Validation and Uncertainty Quantification) means that UQ is a step forward VV but is not included in Validation. Nethertheless, Validation may help UQ.			X	*To meet Requirement 18 form GSR part 4 (Rev.1), comparison of model prediction with experimental data is needed in validation process.*
Canada 57	5.22	Suggest the following changes, *… the development phase, in which the assessment is done by the code developer, and the independent assessment phase, in which the assessment is performed by the code user. Consideration should be given as to whether separate tests must be applied for the validation for the separate phases.*	The two phase approach for validation certainly has merits for complex analyses. Considerations should be given on whether validation exercises must be quarantined between the two phases, and whether there are sufficient independent tests for this purpose.			X	*It is not easy to determine the benefit from this consideration and to implement it, as not many tests are available for complex analyses*
Egypt 3	Para 5.22 page 27	**…in which the assessment is performed by the user code.**	By the user code instead of by the code user, and the same comment at para 5.25			X	*Code user seems better than 'user code'*
Korea 3	§5.23, Second sentence	[errata] (4)…~~nuclear~~ Nuclear power plant	[errata] nuclear -> Nuclear	X			
Canada 68	5.23 item (4)	Nuclear power plant level tests and operational transients. nuclear power	Suggested addition to clarify expectations for new		"…actual nuclear		*To include also other phases and the cold*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		plant level tests are performed on an actual nuclear power plant during, for example, the fuel-in (hot) commissioning phase . Validation through operational transients together with nuclear power plant tests are important means of qualifying the plant model.	designs that such testing will be expected as part of the Commissioning program for the first of a kind prior to commencing to commercial operation.		power plant, for example during the commissioning phase. Validation through operational…"		*phase w/o fuel*
Observer ENISS 39	5.23	(3) Integral effect tests. Integral tests … boundary conditions. In the absence of experimental data, sufficient conservatisms, based for example on code-to-code comparison or bounding engineering judgement, should be allowed to cover the deficiencies on the means to support a full validation. (4) NPP level tests and … qualifying the plant model. In the absence of data, sufficient conservatisms, based for example on code-to-code comparison or bounding engineering judgement, should be allowed to cover the deficiencies on the means to support a full validation.	The sentence in (2) line 4 : "In the absence … full validation" should be common to (2), (3) and (4)		*Last sentence from (2) will be deleted.*		*Better not to add those sentences. §5.23 indicates "The validation should* ***ideally*** *include …". Deviations are not part of §5.23.*
Observer ENISS 40	5.24	Ask for clarification in the document.	We agree with the 5.24 sentence, but there is confusion elsewhere in the document between Generic Validation and Qualification.			X	*No specific suggestion is provided. There is no 'generic' validation; the code is validated only for*

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			Difference should be made between generic validation (first sentence) and specific validation for a specific safety study (second sentence). The first one is related to the validation of the code, the second one is related to the qualification of the code for a safety study.				*the applications for which the validation is performed. Qualification of the code does not apply.*
Observer ENISS 41	5.26	For complex applications, a validation matrix… The validation matrix should be adjusted to the safety case.	The validation must be optimized: not too large, not too small		"5.26 For complex applications …be inaccurate for other data sets. The validation matrix should be adjusted to the application for which the code is validated."		
Japan16	5.29	When performing a validation against experimental data, ~~allowance for errors~~ uncertainty in the measurements should be included in the determination of the uncertainty of the computer code.	Editorial	X			
Observer ENISS 42	5.29/1	Definition of "error" to be added	Glossary		*See Japan-16 and ENISS-3. The term "errors"*		"

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					has been replaced by "uncertainties".		
Observer ENISS 43	5.29 2nd sentence	When performing … of the computer code. In *addition, ~~the evaluation~~ explanations should be provided about the transposition of uncertainties based on scaled experimental results ~~has to be transposed and justified to the uncertainty~~ to the uncertainties relative to the real power plant application*"	Real justification is seldom possible.			X	*It would change the meaning. The term "explanations" may be ambiguous and does not provide 'quantitative assessment'; transposition bias should be evaluated (or conservatisms included) to cover the fact that 'justification is seldom possible'.*
Observer ENISS 44	5.33/1	Replace "range" by "*scope*"				X	*The change doesn't seem to enhance the wording. "Range" is a more quantitative term whereas "scope" has larger meaning, it is not very precise.*
Observer EC-JRC 57	5.33/All	*Remove*	Uncertainty is code-specific but also plant-specific and sequence-specific. Otherwise the entire uncertainty assessment process would be straightforward. Therefore,			X	*To meet Requirement 18 form GSR part 4 (Rev.1), uncertainties of the code should be known through validation process.*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			para 5.33 should be removed.				
Canada 58	5.34	Suggest the following changes, *For a code intended to be conservative regarding certain acceptance criterion, it should be demonstrated that the code prediction ~~bounds~~ is conservative when compared against the experimental data.*	For a code intended to be conservative, it is sufficient to demonstrate that the code predictions are conservative with respect to the experimental. The requirement to demonstrate predictions are bounding is quite onerous and not always attainable.	X			
PL 3.	5.35	Procedures include issues such as the way to compile the input data set, the means of selecting the appropriate models in the code and general rules for preparing the nodalization.	Although nodalization techniques are usually covered by user guidelines more specifically, nevertheless general guidelines for preparing good nodalization should be in the procedure	X			
Japan17	5.35 and 5.36	Para 5.35 and 5.36 should be moved from “VALIDATION OF COMPUTER CODES” to a new part named “NODALIZATION AND USER EFFECT”.	The contents of para 5.35 and 5.36 are not limited to the validation of computer codes.			X	*Nodalization is also part of the validation of the code and cannot be separated from it. If the user does not follow the recommended nodalization (on*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
							which the code is validated) the application is no longer providing reliable results. It seems better to keep both paras in this subsection ('VALIDATION')
Observer EC-JRC 58	5.35, 5.36/All	N/A	According to the rationale of comment 57 on user effects, contents referred in these two paras should be replaced into an independent additional subsection.		*See resolution to Japan-17*	X	
Observer ENISS 45	5.35 and 5.36	Move the paragraphs to another section, as requested in general comment n° 2.	These paragraphs do not fit with the title of the section "validation of computer codes".		*See resolution to Japan-17*	X	
PL 4.	5.36 Line 3	The nodalization should be sufficiently detailed so that all the important phenomena of the scenario and all the important design characteristics of the nuclear power plant analysed are represented. However overcomplicating of nodalization should be avoided as it may have negative impact both on the computational time and the results.	Additional sentence on nodalization – it may appear to the reader that the more detailed and complex nodalization (for example 20 nodes instead of 10) is always welcome, but that is not always the case, and it should be stated in the document.			X	*It seems better not to add the clarification. Computational time is not to be considered; on the other hand this may open the possibility to adopt 'simple' nodalization for the sake of computer time*

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Reviewer: Page....
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Date:</td><td colspan="4">RESOLUTION</td></tr>
<tr><td>Comment No.</td><td>Para/Line No.</td><td>Proposed new text</td><td>Reason</td><td>Accepted</td><td>Accepted, but modified as follows</td><td>Rejected</td><td>Reason for modification/rejection</td></tr>
<tr><td></td><td></td><td></td><td></td><td></td><td></td><td></td><td>whereas the results may be not reliable. It seems also better not to use ‘negative impact’ on the results</td></tr>
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Section 6
DS491. (SSG-2 Rev. 1, Deterministic Safety Analysis for NPPs)

COMMENTS BY REVIEWER Reviewer: ……………………………………………..Page.... of.... Country/Organization: …………………. Date:				RESOLUTION			
Country Org.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
France 17	6.1 Line 4	~~Margins~~ Conservatisms might be introduced in many ways, such as in physical models, in initial and boundary conditions or in acceptance criteria.	Here the word conservatisms should be used instead of 'margins'	X			
Observer ENISS-46	6.2	Please refer to Table 2 in this paragraph	For better understanding, this paragraph should make explicit reference to table 2		*At the end of §6.2 will be added:* "…;see Table 1" *[Note: Table 2 became Table 1]*		
Observer ENISS-47	6.2 line 2	Ask for clarification	"conservative, combined or even best estimate approach, associated with sensitivity analysis": If "associated" is related to "conservative" this is not consistent with #2.9 and #2.10 To be clarified (Table 2)		*The first sentence will be modified as follows:* "Uncertainties in computational predictions…combined or even best estimate approach~~, associated with sensitivity analysis as appropriate~~, or explicitly using …"		
CAN 59	6.3	Suggest the following changes,	The complementary approaches would certainly			X	*§7.30 indicates that acceptance criteria*

		To demonstrate compliance with anticipated operational occurrences acceptance criteria, two complementary approaches should be considered, the realistic approach, using plant control and limitation systems (para 7.17-7.26) and a more conservative approach, using only safety systems (para 7.27-7.44). The acceptance criteria for the conservative approach assuming malfunction of plant control and limitation systems should take into the overall frequency of the postulated event sequence.	demonstrate the robustness of the safety case. The more conservative approach assumes that the plant control and limitation systems do not function as intended. If the frequency of the AOO with control/limitation system malfunction is beyond what is normally considered as the AOO range, then a less stringent acceptance criteria should be applied.				*should be the same for "conservative AOO" and 'realistic' AOO:* "7.30 For conservative analysis of AOO the technical acceptance criteria related to fuel integrity and radiological acceptance criteria should be the same as presented above for realistic analysis of AOO"
GER 23	6.3	6.3. To demonstrate compliance with anticipated operational occurrences acceptance criteria, two complementary approaches should be considered, the realistic approach, using plant control and limitation systems (para 7.17-7.26) and a more conservative approach, using only safety systems (para 7.27-7.44).	The intention of the approach is not clear. From German experience it is only allowed to handle AOOs with operational systems. The usage of safety systems for AOOs is forbidden. Thus, the analyses of AOOs should only consider operational systems available during the transients. The usage of safety systems would contradict the level-of-defense concept. Is the intention of the more conservative approach to show that in case of the failure of operational systems the transition to the DBA level can be managed			X	*§5.75 (e) from SSR-2/1 Rev. 1, indicates to analyse AOO only with safety systems:* "(e) Demonstration that the management of AOO and DBA is possible by safety actions for the automatic actuation of safety systems in combination with prescribed actions by the operator"

			by the plant design? Should be discussed.				
Observer ENISS-48	6.4	Ask for clarification	This paragraph is not consistent with #2.10 ("conservative approach is not suggested") → to be clarified		*§6.4 will be modified as follows:* "6.4. In accordance with SSR-2/1 (Rev.1), §5.26 [1] the deterministic …performed using conservative analysis (see §2.14), including consideration…"		
CAN 18	6.6, 1st sentence	6.6. When best estimate analysis is used, adequate margins to integrity of barriers should still be ensured. It should then be demonstrated by sensitivity analysis that cliff-edge effects[7] ~~(abrupt change in the result of the analysis for a realistic variation of inputs)~~ potentially leading to early or large radioactive releases can be reliably avoided. [7] Definition of a 'cliff-edge effect' is provided in ~~SSR-2/1 (Rev 1), § 5.21 [1]~~ the Safety Glossary. The term ~~„plant parameter"~~ "plant parameter" in the definition should be interpreted in a broad sense, i.e. as any plant physical variable, design aspect, equipment condition, magnitude of a hazard, etc. that can influence equipment or plant performance.	The term "cliff edge effect" is defined in the Safety Glossary. SSR-2/1 does not include the term in its definitions, though it does repeat the text in several footnotes. This guide should not paraphrase that definition in the main text. The application in DSA described in the footnote is sufficient.	X	*Additionally, 6.1 will be modified as follows:* "Margins might be introduced in many ways, such as in acceptance criteria or through conservative assumptions in physical models, and in initial and boundary conditions ~~or in acceptance criteria~~."		
Observer	6.7, 6.8/All	*Please see rationale*	The scope of paras 6.7 and			X	*This is done for DBA*

EC/JRC-59			6.8 regarding sensitivity analysis both in terms of plant state (AOOs, DBAs, DEC) and deterministic safety analysis approach (conservative, BEPU, combined, realistic) should be added. It is the opinion of this reviewer that such activity is restricted to the field of severe accident simulations but only within probabilistic, i.e. Level 2 PRA, analysis.				*and some AOO*
Observer WNA 3	6.7	6.7. For best estimate analysis, parameters to which the analysis results are most sensitive should be identified.	Cliff edge effect is relevant for best estimate analysis, not for conservative analysis			X	*Absence of 'cliff edge effect' has to be always demonstrated*
Observer EC/JRC-60	6.7/7 (addition)	To overcome this issue, global sensitivity analysis techniques should be applied such as Monte-Carlo Filtering, Scatter plots or Sobol indices.	Last sentence in para 6.7 identifies a problem arising from performing sensitivity analysis by varying one parameter at a time yet without offering any solution / recommendation to avoid this shortcoming, which seems to me slightly contradictory.			X	*Out of the scope of this guide; the methods proposed may not yet receive common agreement. The last sentence is a warning suggesting that results should be considered with caution*
CZ 18	6.8 Line 1	6.8. For practical reasons, only a limited number of parameters usually considered to have ~~with~~ the strongest effect on results of analysis can be involved in sensitivity analysis.	Without performing of sensitivity analyses the parameters with strongest effect cannot be exactly identified. Or insert to text reference describing how to evaluate parameters with strongest effect on analyses result.		"6.8. For practical reasons, only a limited number of parameters identified as having the more significant ~~with the strongest~~ effects on results ~~of analysis~~		

					can be involved in sensitivity analysis. Variation…”		
Observer EC/JRC-61	6.9 Line 1	For conservative deterministic safety…	Referred option in bullet 2 is option 3 in Table 2, hence not conservative but BEPU.		*‘Conservative’ is used here according to §2.14. Para 6.9 will be modified:* “6.9. For conservative DSA of AOOs and DBAs (see §2.14), in addition to the fully…”		
Observer ENISS-49	6.9 Line 10	(…) in the second ~~phase~~ case the results are expressed in terms of ranges, percentiles or probability distributions of calculated parameters	When using a BEPU method, the output results may be expressed under various formats: ranges, percentiles (e.g. 95%/95%), probability distribution.		*Calculated parameters hardly follows a known statistical distribution (e.g. Gaussian) this is why the use of ‘Wilks’; suggested change is:* “… in the second ~~phase~~ case the results are expressed in terms of ~~ranges~~ percentiles or confidence intervals of the calculated parameters”		
Observer EC/JRC-62	6.9 Line 11	... in terms of ranges probabilistic distribution functions or confidence	For precision's sake.		*See resolution to ENISS-49*		

		intervals of the calculated parameters.					
Observer EC/JRC-63	6.11/1	…should ~~take into account~~ be updated according to plant real configuration e.g. number of PWR steam generator plugged tubes, implemented plant modifications of any kind affecting modelling components and signals, or any ongoing process such as aging affecting simulated phenomena by the code.	For precision's sake.			X	*Too detailed for the Safety Guide; it seems preferable not to incorporate the change*
Observer EC/JRC-64	6.12/New	Deterministic safety analysis approach in the frame of design extension conditions should consider BEPU approach due to the large uncertainties related to the involved phenomena. Best estimate –default–values provided by the code can significantly deviate from bounding values when uncertainties are incorporated into the calculations. Critical severe accident phenomena such hydrogen generation, corium quenching or fission product release, transport and chemistry feature large uncertainties that can, at least partly, be addressed by identifying governing phenomena, quantifying their uncertainty and propagating through statistical tools by means of representative accident sequence code simulations.	The importance played by uncertainties in severe accident simulation codes has already been discussed. It is not well balanced if the two following subsections are only focused on AOOs and DBAs while not mentioning DECs, moreover when several applications derived from using severe accident codes greatly impact on safety improvements, e.g. mitigating system design such number of PARs or filter type in the Containment Filtered Venting.			X	*The formulation may be considered complex. Nevertheless, the concept is to be covered in Section 7 (see §7.4) where Best Estimate (without BEPU) is allowed.*
Observer EC/JRC-65	6.14 Line 5	… may be different depending on the type of ~~PIE~~ transient	Uncertainty is not (only) PIE-specific but sequence-specific.	X	*(Event sequence could also be used)*		
Observer ENISS-50	6.14	The paragraph should be removed or simplified	For simplification, as these issues are already presented in paragraph 2.11.			X	*See §1.16 (line 2). Section 2 only introduces basic*

							concepts and terminology used in DSA; doesn't provide recommendations ('should' statements), e.g. in §2.11. These recommendations are provided in §6.14
CAN 60	6.15 (To be added at the end)	Suggest the following changes, *Therefore, the appropriate conservatism in initial and boundary conditions should be selected individually, depending on the specific transient and acceptance criteria. Initial conditions that cannot occur at the same time in combination need not be considered.*	Consistent with para 6.19, selection of conservatism for individual initial/boundary conditions should consider if the conditions can occur at the same time.		"…and acceptance criteria. Combinations of initial conditions that cannot occur at the same time do not need to be considered."		
Observer EC/JRC-66	6.15/5 (addition)	… i.e. initial and boundary conditions which are conservative for one specific transient or acceptance criterion could at the same time be not conservative to another transient or acceptance criterion.	For clarification's sake.		*See resolution to CAN-60 (basically included there)*		
Observer ENISS-51	6.20	Operating conditions … negligible ~~probability~~ frequency of occurrence may not need to be considered in selection of conservative initial conditions. Initial conditions should consider stationary state with normal operation equipment operating prior to the initiating fault.	Initial plant state should consider stationary state with normal operation equipment available.	X *(frequency)*	*The last sentence suggested will be added to para §3.5 (now §3.4, once moved down §3.1)*		
Madaga 2	6.21	BEST ESTIMATE DETERMINISTIC SAFETY ANALYSIS WITH QUANTIFICATION OF UNCERTAINTIES FOR	It is better to put the las s of DBAs in LowerCase even within an UpperCase title		*Editorial. When "DBAs" is used in a title of the SG, it will be*		

		ANTICIPATED OPERATIONAL OCCURRENCES AND DBA~~S~~s			*wrote in full* (DESIGN BASIS ACCIDENTS ~~DBAs~~)		
Observer EC/JRC-67	6.21/2	… ~~may~~ should be addressed by in case of making use of best-estimate computer codes in combination with …	For clarification's sake: the text as currently is seems to give to user's choice the alternative of assessing uncertainty in the best-estimate option, i.e. best-estimate code and BICs. But according to option 3 in Table 2, associated uncertainties should indeed be calculated.			X	*BEPU is not the only means thus "may" is the correct term*
PL 6.	Page 34/35 *(General, paras 6.21-6.29)*	General remark about "BEST ESTIMATE DETERMINISTIC SAFETY ANALYSIS WITH QUANTIFICATION OF UNCERTAINTIES FOR ANTICIPATED OPERATIONAL OCCURRENCES AND DBAS" subchapter: statistical method (propagation of input uncertainty) is well described and all important features are discussed. I would like to propose to create further points about "extrapolation of output uncertainty" approach, for the clarification and better understanding. It should cover issues like: • general idea - The inaccuracies are obtained by experimental/calculation comparison, then the inaccuracies is 'extrapolated ' to get uncertainty.	Proposition to expand the information about second method of BEPU analysis - "propagation of output uncertainty". The method is a good alternative to statistical method and more information would be useful.			X	*Not sure whether this relevant change would be supported. It is quite detailed and §6.26 seems more clear and simplified than the proposed text*

		Experimental data are obtained from qualified Integral Test Facilities. • resources and databases of results of calculations and comparisons to experimental data needed to obtain results • Positive like - one broad methodology for uncertainty evaluation, accuracy qualification and answering scaling issue • Expert judgement minimized • drawbacks like the the process of 'extrapolation' of output errors is not based upon fundamental principles					
Observer ENISS-52	6.21 to 6.29	These paragraphs are not specific to AOO or DBA and should be included in section #5 within a subsection related to UQ	For better structure of the document. See general comment nb 2.			X	*This subsection addresses the quantification of margins, so uncertainty quantification is part of Section 6*
France-18	6.23 Line 3.	A reliable assessment of the uncertainties is needed to carry out acceptable best estimate analyses with quantification of uncertainties, especially for the identification of aleatory and epistemic sources of uncertainties, these two different sources should be treated differently when performing the uncertainty analysis. Code-to-data comparisons are the preferred means to quantify the uncertainties. However, a combination	Treatment of aleatory and epistemic uncertainties are different and have to be specified in this document.	X			

		of sensitivity studies, code to code comparisons and expert judgements may also be used as an input for the assessment					
CAN 61	6.23 Line 3 plus line 5	Suggest the following changes, *Code-to-data comparisons are the preferred means to quantify the epistemic uncertainties. However, a combination of sensitivity studies, code to code comparisons and expert judgements may also be used as an input for the assessment. For aleatory uncertainties, the preferred means is the collection of nuclear power plant data of initial and boundary conditions that are relevant to the events being considered.*	As noted in this para, it is important to recognize the distinction between aleatory and epistemic uncertainties. This is particularly important for some applications or methods of Best Estimate Analysis with Uncertainties. Aleatory uncertainties generally refer to random variations in process conditions while epistemic uncertainties are related to ability to measure or predict a condition accurately. Use of code-to-data or code-to-code comparisons cannot readily establish the aleatory uncertainties.	X	*Reference to GSR Part 4 (Rev. 1), Req. 17 [2] will be made at the end of the existing wording of §6.21.*		
USA 6	6.23 & 6.24 (p. 34)	Remove line between 6.23 & 6.24.	Editorial. Line serves no purpose.	X	Editorial		
USA 7	6.28 (p. 35), Last sentence	However, attention should be given to the fact that the regression or correlation techniques might ~~have~~ also have drawbacks, especially when the response is not linear or when the cross-correlation effects are important.	Editorial / clarity	X	Editorial		
Observer EC/JRC-68	6.29/3	… that is analyzed. The ranking PIRT tool application should identify…	For precision's sake.		"... for each event that is analysed. This PIRT ~~The~~		

					~~ranking~~ should identify the most important …”		
Observer EC/JRC-69	6.29/5 (addition)	… on available data. If the number of output relevant phenomena is high, an additional filter taking only those lacking on sufficient knowledge might be applied.	Several international PIRT applications have performed this further filtering step.			X	*It seems not necessary, quite detailed.*
Observer EC/JRC-70	6.29/6	… to determine the overall uncertainty of the figures of merit used to check compliance with acceptance criteria specific of that particular code, plant characterization and accident sequence simulation.	It is unclear what is the reference subject when talking about 'the same process can be applied'. What is the mentioned process?			X	*It seems not necessary, quite detailed.*
PL 5	6.29 Line 8	Proposition of additional text: High level of expertise and experience is needed to fix ranges of variations of input parameters and to carry out PIRT process.	PIRT process is very sensible to expert judgment so it should be noted that expertise and experience is needed.			X	*It seems not necessary. The idea is covered by the first sentence of 6.29:* “… based on expert judgement…”

Resolution to Comments on Section 7
DS491 Step 7: Deterministic Safety Analysis for NPPs

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Japan 18	Section 7	In section 7, it is implied that the Option 4 is used in the Realistic AOO and the Option 2 and 3 are used in the rest of analyses. There should be explicit guidance on which option should be used in each type of analysis.	Clarification				*Realistic approach should be used also for severe accident analysis. In any case paras 2.8 to 2.15 indicate options to perform DSA in a wide range of purposes, not directly and exclusively linked and limited to each plant state. It is understood that "strong recommendations" on which approach should be used for scenarios under examination should not be made in this SG.* *See additional elements in the resolution to Germany-24*

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Germany 24	Section 7, Pages 35 to 42	-	General Comment: The intention regarding the selected structure of Chapter 7 is unclear. For AOOs both conservative and best estimate approaches are discussed. For DBA only the conservative approach is treated. The best estimate approach for DBA is missing. Should be added. The structure of chapter 7 should be made more clear (improvement of the order of the sections)		*A reference to SSR-2/1 (Rev.1) § 5.26 will be added in §7.27:* "7.27. Realistic analysis for DBA is not permitted; one of the conservative methods[8] (Options 1, 2 or 3 from Table 1) should be used. The conservative analysis for AOO and DBA should…". *The footnote will be updated.* *(See resolution to Japan 18 too)*		
Czech 19	7.5	Evaluation of the source term should thus involve determining the behaviour of the radioactive species along this route up to their release to the environment ~~release to the atmosphere~~.	Text clarity. Release can be not only to the atmosphere but to hydrosphere too.	X			
Observer EC/JRC 71	7.7 Line 3	… occurrences and DBAs ~~and design extension conditions~~.	Independently on whether agree or not with integrating dedicated DEC-related systems (e.g. PARs, containment flooding, etc.) in plant limits and		*Initial conditions of reactor power, coolant inventory etc. will be important for DEC analysis.*		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			conditions, current IAEA NS-G-2.2 does not include them and neither existing collections of plant limits and conditions.		*The para will be clarified as follows:* "7.7. The limits and conditions used in normal operation, such as reactor power and coolant inventory, should cover all important…"		
Canada 19	7.8	7.8. All ~~possible~~ operating modes of normal operation covered by operational limits and conditions should be analysed, with particular attention paid to transient operational regimes such as changes in reactor power, reactor shutdown from power operation, reactor cooling down, handling of irradiated fuel and off-loading of irradiated fuel from the reactor to the spent fuel pool.	"All possible" seems excessive. Many modes are foreseen at the design and construction phase, and limits and conditions are set for them. But this is far short of "all possible" modes. Some unusual modes will be defined if needed and the analysis performed to justify them. They will not be part of the standard set documented in the OLCs.	X			
Observer EC/JRC 72	7.10 Line 4 (addition)	… be avoided in the entire spectrum of transients belonging to the normal operational plant state as defined by the operational limits and conditions and considering the entire plant operating states from full power to shutdown conditions. Transitions from	For clarification's sake.		*Used "operating modes"* "…avoided in all the transients, as defined by the operational limits		"… be avoided ~~in the entire spectrum~~ in all the transients, ~~belonging to the normal operational plant state~~ as defined by the operational

Reviewer: Country/Organization:		COMMENTS BY REVIEWER Page.... of.... Date: 25/05/16		RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		one operating state to another as anticipated according to operational guidelines should be also taken into account.			and conditions, and considering all the operating modes. Transitions from one operational state to another, as anticipated according to operational guidelines (??), should be also taken into account		limits and conditions, and considering ~~the entire plant~~ all the operating modes~~from full power to shutdown conditions~~. Transitions from one ~~operating~~ operational state to another, as anticipated according to operational guidelines, should be also taken into account.”
Czech 20	7.11 Last sentence	However, demonstration of compliance with the radiological acceptance criteria for normal operation is not covered by this Safety Guide.	Completing the reference of relevant Guide is recommended.	X	“,,, However, compliance with the radiological acceptance criteria [3] is not covered by this Safety Guide.		
Observer ENISS-53	7.12	7.12. Systems credited in deterministic analysis of normal operation should be limited to normal operation systems, including plant control systems. No other plant systems should be actuated or be affected (especially the availability of safety-related SSCs) during transient normal operational modes.	For completeness			X	*The clarification seems not necessary*
Madagas 3	7.14	~~I&C~~ shall be replaced	I&C shall be replaced by its ***right meaning***. As it can be	X	“including instrumentation		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			"Information and Communication", "Installation & Commissioning", "Instrumentation and Communication", …		and control ~~I&C~~ and mechanical…".		
Canada 20	7.17 1st sentence, line 2	7.17. The main objective of the realistic analysis of anticipated operational occurrences is to check that the plant operational systems (in particular control and limitation systems) can prevent most anticipated operational occurrences from evolving into accident conditions and that the plant can return to normal operation following an anticipated operational occurrences.	"Most" should be added as indicated. The control and limitation systems cannot control all AOOs. This is clear in 7.18. See also SSR-2/1 para 2.13 (3) and para 5.75, item (e). Clearly, there is no expectation that control systems must deal with all AOOs.		7.17. The main objective … systems) can prevent a wide range of anticipated operational occurrences …"		
Observer ENISS-54	7.18 Line 2	7.18. For many PIEs the control and limitation systems in combination with inherent plant characteristics <u>and operator actions following normal or abnormal operation procedures</u> will compensate (…)	In addition to system and plant features, operator actions, following normal or abnormal procedures, may be needed.		7.18. For many PIEs the control and limitation … inherent plant characteristics <u>and operator actions</u> will compensate for the…"		
Observer EC/JRC 73	7.18/2,7.20/4	*Read rationale*	In both paras, AOOs are defined as transients beyond normal operation but without leading to reactor trip and safety systems			X	*The text does not cover all the range of AOOs. There are some that must be dealt with by safety*

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			actuation. However, such statement does not belong to AOOs whereas a typical instance of such transients is LOOP where automatic reactor trip is expected to occur. Please update if necessary.				*systems*
Observer ENISS-55	7.19 Line 2	…It is therefore advisable to demonstrate by the analysis that, in case of the operation of the plant control and limitation systems as intended, the safety systems are no unnecessarily initiated <u>and, if their initiation is necessary and unavoidable, the initiation of safety systems will not markedly increase the risk that the anticipated operational occurrence is escalated into an accident.</u>	The reactor trip (scram) function is necessary in some DBC2 events, for example, loss of turbine condenser in BWRs, and cannot be safely avoided in these cases. In addition, 3.41 explicitly considers that in some DBC2 events, a scram is necessary, as it requires the postulations of ATWS cases, and 7.20 also allows reactor trip in cases where unavoidable. Our proposal also agrees with the content of 7.33.		*The sentence was verified rejecting the comment. An editorial correction was identified*: 7.19. In addition, the anticipated operational …that, in case of the operation of ~~the~~ plant control and limitation systems …”	X	*The suggested sentence may be confusing; it seems better not to include it.*

Reviewer: Country/Organization:		COMMENTS BY REVIEWER Page.... of.... Date: 25/05/16		RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Belgium 4	7.22	Delete specifications on percentage? (95% probability; 95% confidence; 10-15%). Or include a flexibility statement?	Art. 7.22 seems to us the only article with such precise prescriptions. Article 6.24 also gives %-values, but that article includes some flexibility statement. Make also 7.22 somewhat more flexible?			X	*Sentence states "typically" so the flexibility is already included.*
Japan 19	7.26	This paragraph should provide specific guidance on analysis assumptions and treatment of uncertainties for the realistic AOOs.	Clarification	X	The following text will be added at the end of 7.26: "… determination of the PIEs. Normally, uncertainties are not considered in realistic analysis of AOO. For operational considerations (such as plant reliability), treatment of uncertainties may be applied to the control and limitation systems."		
France 19	7.26 Title	**Analysis assumptions ~~and treatment of uncertainties~~**	No mention is given about treatment of uncertainties		*See resolution to Japan-19*		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			and this topic should be deleted from the title of the paragraph				
Germany 25	7.27	7.27. Conservative analysis[8] of anticipated operational occurrences and DBAs should demonstrate that the safety systems alone are capable of fulfilling the following safety requirements	The AOOs should only be handled by operational systems. The usage of safety systems should only be allowed for DBA and design extension conditions (see also comment 23). The conservatism regarding AOOs should be considered e. g. by unfavorable initial and boundary conditions.		*(Final wording of 7.27 according to Germany-24 and ENISS-56)*	X	*As an example, most NPPs rely on scram to protect against loss of all main coolant pumps. This event is typically in AOO frequency range.* *Also, see SSR-2/1 para 4.11 (d), 4.13 and para 5.75 (e).*
Observer ENISS-56	7.27	(…) should demonstrate that the safety systems ~~alone~~ and the operator actions following EOPs are capable of fulfilling (…)	Operator actions, in addition to safety systems, are most often required.	X	*To better align with SSR-2/1 §5.24, the §7.27 will be modified as follows:.* "… should demonstrate that the safety systems alone in the short term, and with operator actions in the long term, are capable of achieving a safe state by fulfilling, the following safety		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					requirements …"		
Observer ENISS-57	7.27	Include here "safe state" from SSR-2/1	SRR2-1 Req. 19 §5.24 that requires a safe state to be reach and maintened for DBA should be added.	X	*(Final wording of 7.27 according to Germany-24 and ENISS-56)*		
Germany 26	7.28	7.28. The safety analysis should demonstrate that the acceptance criteria relevant to the event are met. In particular, it should be demonstrated that some or all of the barriers to the release of radioactive material from the plant will maintain their integrity to the extent required.	The German understanding is that all barriers have to maintain for the AOOs. That is reflected by the set of acceptance criteria used for that level of defense. For DBA in maximum two barriers (fuel matrix and fuel rod cladding) of a limited number of rods are allowed to fail. Modification of the formulation of the sentence?			X	*It may not be possible to maintain all barriers for all AOO. For example, SG tube leakage is a failure of one of the barriers and bypassing another as an initiating event.* *This para is generic. There is a specific provision in 7.30 which requires meeting 7.20 (which deals with integrity of barriers).*
Observer ENISS-58	7.29	7.29. The safety analysis should establish the design capabilities, safety system set points, EOPs to ensure that the fundamental (…)	Operator actions, in addition to safety systems, are most often required. AOOs and DBAs analysis support EOPs definition.	X	*According to the resolution to ENISS-56, it will be modified as follows:* "7.29. The safety analysis should establish the		

Reviewer: Country/Organization:		COMMENTS BY REVIEWER Page.... of.... Date: 25/05/16		RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					design capabilities, safety system set points, and operating procedures to ensure that the fundamental …”		
Observer EC/JRC 74	7.31	*Please see rationale*	The scope in para 7.31 should be indicated. Apparently it only refers to DBAs but lacks of indication.			X	*7.31 belong to the subsection* “Conservative Analysis for AOOs and DBAs”; *see heading before 7.27.*
Japan 20	7.32.	7.32. Specific decoupling criteria should be defined in order to prove that the three ~~main~~ safety functions can be ensured	Clarification. According to the IAEA glossary, “main safety functions” means “fundamental safety functions”. Consider deleting “main” to avoid confusion.	X	*The term “fundamental” will be used. (Note for convenience: In resolution to Japan-10 (about §3.51) it is indicated:* *A foot note will be added:* (*) According to the IAEA Safety Glossary (2016) the term “main safety functions” is equivalent)		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Germany 27	7.32	7.32. Specific decoupling criteria should be defined in order to prove that the three main safety functions can be ensured in any condition and that, in an anticipated operational occurrences or DBA, at least one safety barrier remains able to limit the radiological releases to the environment.	For AOOs and DBAs the requirements should be that more than one barrier will be intact. Modification of the wording?	X	"7.32. Specific decoupling criteria should be defined in order … condition and that, in an AOO or DBA, some or all of the barriers are able ~~at least one safety barrier remains~~ to limit the radiological releases …"		
Observer ENISS-59	7.32 Last line	(…) ~~at least one safety barrier remains able to limit the radiological releases to the environment~~ barriers to the release of radioactive material from the plant will maintain their integrity to the extent required to meet Req. 4.10.	Proposal		*See resolution to Germany-27*		
Observer ENISS-60	7.33 1st bullet	(…) ~~and a DBA (in combination with a single failure) should not generate design extension conditions~~.	As single failure is part of the DBA analysis, a DBA PIE + single failure makes the DBA conditions. It can not be a DEC condition.			X	
Finland-3	7.33 3rd bullet Line 3	… Systems used for accident mitigation should be designed to withstand the maximum loads, stresses and environmental conditions for the accidents analysed. This should be assessed by separate analyses covering environmental conditions *and ageing*	Add: Ageing Ageing should be considered with the assessment of the environmental conditions. The equipment/SSCs should be able to perform their	X			

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		(i.e. temperature, humidity or chemical environment) and thermal and mechanical loads on plant structures and components. The margins considered in the design should be commensurate with the probability of the loads to be considered. …	intended function even at the end of their lifetime.				
Observer ENISS-61	7.33 3rd bullet Line 3	(…) i.e. temperature, humidity, <u>irradiation</u> or chemical environment)	Proposal		"…humidity, radiation or chemical environment…"		
Observer ENISS-62	7.33 5th bullet Last line	The number of fuel cladding failures which could occur should be limited for each type of PIE to allow the global radiological criteria to be met <u>and to allow decoupling hypothesis retained to define equipment qualification requirements to be met.</u>	The number of cladding failures should also be consistent with the decoupling hypothesis that may have been retained to define qualification requirements for SSCs.	X	*At the end of 5th bullet it will be added:* "… the global radiological criteria to be met <u>and also to limit the level of radiation used for equipment qualification."</u>		*(See resolution to Germany-28 clarifying that §7.33 applies to DBA)*
Germany 28	7.33 6th bullet, Page 40	— … —In <u>DBAs</u> ~~accidents~~ with fuel uncovering and heatup, a coolable geometry and structural integrity of the fuel rods should be maintained. — …	The relevant group of events for that requirement should be made clearer.	X	*To clarify that §7.30 relates to conservative AOO and §7.33 to DBA, line 1 of §7.33 will be modified:* "7.33. The detailed		

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					acceptance criteria for DBA should typically include …”		
Czech 21	7.33 7th bullet Line 1	No event should cause the temperature~~,~~ ~~pressure~~ or pressure differences between containment compartments to exceed values which have been used as the containment design basis.	Text clarity.			X	*Need both “pressure” and “pressure difference”*
Japan 21	7.33, 8th bullet	— Subcriticality of nuclear fuel in reactor after shutdown, in fresh fuel storage and in the spent fuel pool should be maintained. Temporary recriticality* may be acceptable for certain events and plant operating modes, however without exceeding criteria associated with sufficient cooling of the fuel. Footnote: In case of steamline break for PWR.	Clarification. If the “Temporary recriticality” in 8th bullet is related to steamline break for a PWR plant, such clarification or limitation is needed.	X	“- Temporary recriticality (e.g., steam line break in PWR) may be acceptable for certain …”		
Germany 29	7.33 9th bullet Page 41	—There should be no initiation of a brittle fracture or ductile failure from a postulated defect of the reactor pressure vessel (RPV) during the plant design life for the whole set of transients and postulated DBAs ~~accidents~~. — …	The relevant group of events for that requirement should be made clearer.	X	—There should be no initiation … the plant design life for the whole set of ~~transients and~~ postulated DBAs ~~accidents~~. *Last bullet will be*		

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					modified accordingly: "… dynamic loads during ~~transients and during~~ DBAs so that safe…"		
Observer ENISS-63	7.35	- For DBAs : - Normal operation systems that are in operation at the beginning of the event ant that are not affected by the initiating event and the consequences of the PIE, can be assumed to continue to operate. - Safety systems designed […]	Crediting systems in service should also be applied to DBA in addition to AOO events.	X	See change for Canada 7.35		
Canada 21	7.35	7.35. The conservative considerations regarding the availability of plant systems should typically include the following: — ~~For anticipated operational occurrences,~~ Normal operation systems that are in operation at the beginning of the event and that are not affected by the initiating event and the consequences of the PIE~~,~~ can be assumed to continue to operate. ~~— For DBAs:~~	All these bullets apply to DBA and to conservative AOO analysis for demonstrating the effectiveness of the safety systems. See SSR-2/1 para 5.75 item (e) "*5.75. The deterministic safety analysis shall mainly provide:* *(a)…(d)* *(e) Demonstration that the*	X			

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		- Safety systems […] should be assumed to operate with conservative performances. - Any control or limitation systems should be assumed to start operating only if their functioning would aggravate the effects […]. - A single component failure should be assumed to occur in the operation of the safety groups required for the initiating event, in addition to the initiating failure and any consequential failures (the Single Failure Criterion). If the single failure is applied to the reactor scram system, the insertion of the control rod that has the greatest effect on reactivity should be assumed to fail. - Safety features for DEC should not be credited in the analysis.	*management of anticipated operational occurrences and design basis accidents is possible by safety actions for the automatic actuation of safety systems in combination with prescribed actions by the operator;*" Effectively, an [AOO + failure of the control and limitation function] can be considered to be a DBA. It I can be seen as a multiple failure event in the DBA frequency range. Also clarify that the last bullet is the Single Failure Criterion which is well described elsewhere. Final bullet from earlier draft seems to have been lost and should be restored.				

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Observer WNA 4	7.35	For anticipated operational occurrences, normal operation systems that are in operation at the beginning of the event and that are not affected by the initiating event and the consequences of the PIE, can be assumed to continue to operate steadily	To be specified in order not to contradict the following bullet regarding control & limitation systems. Basically the aim is to keep main coolant pumps operation for instance (steady operation, no control associated) and to consider normal controls "frozen"		*Covered by resolution to Canada-21*		
Germany 30	7.35	- Single failure should be assumed to occur in the operation of the safety systems ~~groups~~ required for the initiating event, in addition to the initiating failure and any consequential failures. Dependent on the selected acceptance criterion the single failure should be put to a system/component leading to the largest challenge for the safety systems. If the single failure is applied to the reactor scram system, the insertion of the control rod that has the greatest effect on reactivity should be assumed to fail.	Single failures are only postulated for safety systems. It should be mentioned where to put a single failure in order to reach the worst initial and boundary condition for the analysis.	X *2nd change*		X *First change*	*First change:* *See SSR-2/1 Req. 25, §5.39 using "safety groups" for single failure criterion*
Japan 22	7.37.	7.37. For conservative safety analysis, credit should not be taken for operator ~~diagnosis of the event and starting the actions, typically earlier than in 30 minutes if performed in the control room, or 60 minutes for the field actions.~~ Action to limit the evolution of a design basis accident within a	Take off specific values and keep the original sentence (SSG-2, 4.10) to be more general expression.	X *First part*	*First part: treated in common with the proposal from Canada-62.* 7.37. For conservative safety … operator diagnosis of the	X *Second part*	"Exceptionally, the design may take credit for earlier operator action …" *According to current practices in the preparation of Safety Satandards it seems*

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		specified time. Exceptionally, the design may take credit for earlier operator action, but in these cases the actuation times should be conservative and should be fully justified. Conservative assumptions should be made with respect to the timing of operator actions. It should be assumed that in most cases post-accident recovery actions would be taken by the operator.			event and starting the actions. The corresponding time claimed should be justified and validated for each specific reactor design; for example earlier than in 30 minutes for operator diagnosis …”		*better not to include this exception*
Canada 62	7.37	Suggest the following changes, 7.37. For conservative safety analysis, credit should not be taken for operator diagnosis of the event and starting the actions, typically earlier than in ~~30~~ 15 minutes if performed in the control room, or ~~60~~ 30 minutes for the field actions. The timing should be justified and validated for specific reactor design.	The proposed credit for operator is more stringent than current practice for PHWR. The ability to complete the operator action should be justified and validated for each reactor design.		*Treated in common with the proposal from Japan-22; see resolution.* *(Note: Figures in Canada for illustration: 30’ and 60’for new NPPs (REGDOC-2.5.2). Existing NPPs can use 15’ and 30’ (REGDOC-2.4.1))*		
Observer WNA 5	7.42	7.42. If a conservative or combined methodology is applied …	According to wording defined in table 2	X			
Canada 22	7.43 First	7.43. In addition to the postulated initiating event itself, a loss of off-site	Loss of offsite power is an over-conservative	X			

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	sentence	power ~~should~~ may be considered as additional conservative assumption. If LOOP ~~should be~~ is considered as an additional failure ~~occurring~~ it may be assumed to occur at a time which has the most negative effect regarding the barrier integrity.~~, then s~~ Some acceptance criteria should be adapted taking into account the probability of this combination.	assumption for shutdown modes. Text should not require LOOP for all DBAs.				
Germany 31	7.43	7.43. In addition to the postulated initiating event itself, <u>for DBAs</u> a loss of off-site power should be considered as additional conservative assumption. LOOP should be considered as an additional failure occurring at a time which has the most negative effect regarding the barrier integrity, then some acceptance criteria should be adapted taking into account the probability of this combination.	Should the superposition of initiating events with the LOOP be limited to DBAs? That seems to be common practices.		*See Canada 22 about this para.*		
Observer ENISS-64	7.43	In addition to the postulated initiating event itself, a loss of off-site power should be considered as additional conservative assumption. ~~LOOP should be considered as an additional failure occurring at a time which has the most negative effect regarding the barrier integrity,.~~ Then some acceptance criteria should be adapted taking into account the probability of	The LOOP superimposition rule should be considered as a conventional rule bringing robustness to the safety demonstration but its origin is still not shared internationally. As such, it is difficult to define at this stage, for example at which time it should be applied.		*See Canada 22 about this para*		

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		this combination.	As such, these conditions of application should rather be debated with national authorities.				
Germany 32	Section 7, Page 42	-	A chapter regarding detailed deterministic analyses for DBA is missing. See also general comment # 24			X	*The level of detail seems compatible with other paragraphs and there is additional information in other sections.*
Japan 23	7.45. 7.55.	7.45. ... adequate margin to avoid cliff-edge effects. 7.55. ... adequate margin to avoid cliff-edge effects.	Editorial.	X			
Canada 23	7.46	7.46. Acceptance criteria for design extension conditions should meet the requirement of SSR-2/1 (Rev. 1) §5.31A [1].The same or similar technical and radiological criteria as those for DBAs ~~should~~ may be considered for these conditions to the extent practicable.	Para 7.46 exceeds the requirements of SSR-2/1. The radiological criteria do not have to be the same for DBA and DEC. SSR-2/1 para 5.25 says DBA should "*have no, or only minor, radiological consequences, on or off the site, and do not necessitate any off-site protective*	X			

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			actions” SSR-2.1 para 5.31A says DEC should need only “*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*”				
Switzerland 7	7.46	The same or similar technical ~~and radiological~~ criteria as those for DBAs should be considered for these conditions to the extent practicable. Radioactive releases shall be minimized as far as reasonably	We request to change it in accordance with the WENRA-RL F4.14 and SSG-2/1. This is in terms of the Graded Approach.	X *Second part*	*Second part; it will be added:* “…to the extent practicable. Radioactive releases should be minimized as far as reasonably practicable.”	X *First part*	*It seems to exceed the requirements of SSR-2/1. See resolution to Canada23*

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Japan-24	7.48 Second sentence	Special attention should be paid to the frontline systems (e.g., sump screen blockage) and support systems (electrical, ventilation, cooling,) when assessing the independence of safety systems regarding the postulated failures (e.g., internal-flooding).	Clarification and addition of examples. Sump screen blockage problem is important for long-term cooling during and after SA condition.		"Special attention should be paid to other factors affecting safety systems (e.g., sump screen blockage) and support systems (electrical, ventilation, cooling,) when assessing the independence of safety systems regarding the postulated failures (e.g., internal-flooding)		*Better not to incorporate a new class of systems (frontline).*
Observer ENISS-65	7.49	Please add: If, for some events, normal operation or limitation systems are considered as available, it should be ensured that these are not lost in the PIE, and the PIE group represented by the analysis should be selected accordingly.	Provided that normal operation systems including control and limitation systems are not affected by the PIE and its consequences, and when relevant, the failures that define the DEC condition, they should be considered available to be credited. In some countries, some normal operation systems are allowed to be credited as available, if the PIE does not			X	*For safety demonstration purposes the safety features for DEC are the SSC to be credited*

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			affect them (for example, by crediting normal AC power supply systems in the analysis of loss of seawater, as the likely reason for the loss is an oil spill or similar event that has no effect on that system).				
Belgium 5	7.50 and 7.56	Delete one of these articles	These two articles are saying the same.	X	*7.50 has been removed*		
Observer ENISS-66	7.50	Please remove	Redundant with 7.56	X			
Germany 33	7.50	7.50. The single failure criterion need not be applied in the analysis of design extension conditions without significant fuel degradation. <u>Furthermore, no additional failure of a system/component due to maintenance has to be considered.</u>	For clarification it should be mentioned that also no additional failure of a system/component due to maintenance has to be considered.	X	It is in contradiction to the realistic approach. It will be added: Furthermore, no additional failure of a system <u>or</u> component due to maintenance <u>should</u> be considered. *According to Belgium-5 and ENISS-66, it will be added to 7.56*		

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Japan-26	7.51.	7.51 Non-permanent ~~systems and~~ equipment should not be considered for demonstration of adequacy of the nuclear power plant design…	Editorial. To be consistent with SSR-2/1 (Rev. 1). "Non-permanent systems" is not used in SSR-2/1 (Rev. 1).	X			
Finland-4	7.51	7.51. Non-permanent systems and equipment should not be considered for demonstration of adequacy of the nuclear power plant design. ~~Such equipment is typically considered to operate for long-term sequence and is considered available in the development of emergency operating procedures or accident management guidelines~~.	Unnecessary and ambiguous sentence. There is no need to say here, when non-permanent systems are operating, if they should not be taken into account in DECs without core melt.		*See resolution to Canada-24*		
Canada 24	7.51	7.51. Non-permanent systems and equipment should not be considered for demonstration of adequacy of the nuclear power plant in the short term. Such equipment is typically considered to operate for long-term sequence and is considered available in the ~~development of~~ accordance with emergency operating procedures or accident management guidelines. Non-permanent equipment may be credited after 8 hours for equipment stored on site or 72 hours for equipment stored off site. The time claimed should be	Some modern designs have such long passive cooling capability that non-permanent systems are perfectly acceptable. It would be better to set a time limit after which non-permanent equipment may be credited. This is analogous to the operator action time rules in para 7.37.	X *First and second changes*	*The last two sentences will be modified as follows:* "…management guidelines. The time claimed for availability of non-permanent equipment should be justified; for example, for new nuclear power plants, the safety analysis may		

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		justified.			credit non-permanent equipment after 8 hours for equipment stored on site or 72 hours for equipment stored off site."		
Germany 34	7.51	7.51. Non-permanent systems and equipment should not be considered for demonstration of adequacy of the nuclear power plant design. Such equipment is typically considered to operate for long-term sequence and is considered available in the development of emergency operating procedures or accident management guidelines.	Mobile equipment is also used for preventive measures, like a mobile pump for secondary side feeding of steam generator. Their effectiveness is also shown by deterministic event analyses. Preventive measures by portable equipment should not be excluded here by definition.		*See resolution to Canada-24*		
Switzerland 8	7.51	*Non-permanent systems an equipment should not be considered for demonstration of adequacy of nuclear power plant design. Such equipment is typically considered to operate for long-term sequence and is considered available in the development of emergency operation procedures or accident management guidelines.*	For new plants this can be a clear design requirement but are from our understanding in contradiction with the requirements for DBA's. E. G. SSG-2/1 Ziff. 5.11-5.15 allows already for DBA's the use of mobile equipment *(5.15 Any equipment that is necessary for actions to be taken in manual response and recovery processes shall*		*See resolution to Canada-24*		

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			be placed at the most suitable location to ensure its availability at the time of need and to allow safe access to it under the environmental conditions anticipated.) Also 5.28 and 5.29 are focusing on all other items important to safety or features that are designed for use in, or that are capable of preventing or mitigating…..which not explicitly exclude mobile equipment. We request to cancel this requirement or complete rewrite it. Normally even for new plants (DEC’s) AM-Guidelines or mobile equipment (if available) will focus on measure to prevent significant fuel degradation if sufficient time is available. This clearly also meats to cope with DECs for existing plant (see WENRA-RL F4.3).				

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Observer EC/JRC 75	7.51/2	*Please see rationale*	Whole second sentence providing rationale for not accounting for non-permanent systems is unclear. Two arguments are provided: the 'long-term argument' might be better explained, maybe by referring to the time needed to actuate such flexible systems that go beyond to DEC times. The 'EOP and SAMG argument' sounds contradictory: precisely because those systems are accounted for in EOP (just like any other safety system), they should be taken into account in the safety analysis accordingly. Therefore, first argument should be better explained and second argument removed unless clarified.		*See resolution to Canada-24*		
France 16	7.51	Non-permanent systems and equipment should not be considered for demonstration of adequacy of the nuclear power plant design in the short term phase of an accident. Such equipment is typically considered to operate for long-term sequence and is considered available in the	Mobile equipment should be allowed for long term plant stabilization.		*See resolution to Canada-24*		

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		development of emergency operating procedures or accident management guidelines.					
Canada 25	7.52	7.52. Best estimate assumptions can be used for the analysis of design extension conditions. Conservative assumptions as described for DBAs ~~should~~ may be used to the extent practicable. ~~A more realistic approach that considers the information available and the inherent uncertainties in the data might be acceptable but should also consider the additional challenges of design extension conditions.~~	This paragraph exceeds the requirements of SSR-2/1. SSR-2/1 does not use the word "conservative" anywhere under Requirement 20 for DEC. This should be "Best-estimate assumptions" in keeping with the "engineering judgement" and "practicable provisions" wording used in SSR-2/1. Requirement 20. Also 5.27, "best-estimate analysis" and "to the extent practicable". See also SSR-2/1 para 5.75 item (f) "*5.75. The deterministic safety analysis shall mainly provide:* *(a)…(e)* *(f) Demonstration that the management of design extension conditions is possible by the automatic*	X	*Treated with GER-35. Changed to:* 7.52. Best estimate assumptions should be used for the analysis of design extension conditions. Conservative assumptions as described for DBAs ~~should~~ may be used to the extent practicable. ~~A more realistic approach that considers the information available and the inherent uncertainties in the data might be acceptable but should also consider the additional challenges of~~		

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			actuation of safety systems and the use of safety features in combination with expected actions by the operator.”		~~design extension conditions.~~		
Germany 35	7.52	7.52. Conservative assumptions as described for DBAs should be used to the extent practicable. A more realistic approach that considers the information available and the inherent uncertainties in the data might be acceptable but should also consider the additional challenges of design extension conditions.	The best-estimate approach should be used for design extension without significant fuel degradation. Those analyses e. g. are performed for showing the effectiveness of preventive EOPs.	X	See resolution to Canada-25		
Canada 26	7.53	7.53. Since the physical phenomena taking place in design extension conditions without significant fuel degradation do not qualitatively differ from those present in DBAs, the requirements on the selection, validation and use of computer codes specified for DBAs should also apply in principle for analysis of design extension conditions without significant fuel degradation, though a lower level of confidence is acceptable.	Again, this exceeds the requirements of SSR-2/1. Best estimate analysis can be used. See comments on 7.46, 7.51 and 7.52.			X	*A lower level of confidence is not defined.*
Observer ENISS-67	7.53	~~Since the (…) the~~ The requirements (…)	It is not because physical phenomena are the same between DBAs and DECs	X			

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			that computer code used for DECs should be validated.				
Canada 27	7.55	7.55. When best estimate analysis is performed, margins to the cliff-edge effect should be ~~proved~~ shown by sensitivity analysis demonstrating to the extent practicable that, when more conservative assumptions are considered for dominant parameters, there are still margins to the loss of integrity of physical barriers.	Again, this exceeds the requirements of SSR-2/1. Requirement 20, paras 5.27 to 5.31A do not mention "margins" or "cliff edge effects".		7.55. When best estimate analysis is performed, margins to avoid the cliff-edge effect should be ~~proved~~ shown, for example by sensitivity analysis demonstrating to the extent practicable that, when more conservative …"		
Observer WNA 6	7.56	To be deleted	Already specified in 7.50			X	*7.50 was deleted instead*
Poland 7	7.56	Proposition to delete this point as this is the repetition of 7.50	Repetition			X	*7.50 was deleted instead*
Germany 36	7.56	~~7.56. For design extension conditions without significant fuel degradation, single failure criterion does not need to be applied.~~	That is a repetition (see also 7.50). Should be deleted here.			X	*7.50 was deleted instead*
Japan-25	7.56	Delete 7.56.	Redundant with 7.50.			X	*7.50 was deleted instead*
Canada 28	7.56	7.56. For design extension conditions without significant fuel degradation, single failure criterion does not need to be applied and unavailability due to	Make it clear that the requirement on safety systems in para 7.36 of the guide does not apply in	(X)	*Covered in Germany-33*		

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		maintenance does not need to be considered.	DEC.				
Canada 29	7.57	7.57. ~~The~~ From the best estimate analysis of severe accidents, ~~should identify the most severe~~ bounding plant parameters ~~resulting from the core melt sequences~~ should be identified, and ~~demonstrate~~ it should be demonstrated that: […]	This exceeds the requirements of SSR-2/1. Requirement 20 for DEC. SSR-2/1 3.27, last sentence states, "*The effectiveness of provisions to ensure the functionality of the containment could be <u>analysed on the basis of the best estimate approach</u>*."		7.57. The analysis of severe accidents should identify the ~~most severe~~ bounding plant parameters resulting from the postulated core melt sequences, and demonstrate that: (…)		
Germany 37	7.57	7.57. The analysis of severe accidents should identify the most severe plant parameters resulting from the core melt sequences, and demonstrate that: - the plant can be brought into a state where the containment functions can be maintained in the long term - the plant structures, systems, and components (e.g., the containment design) are capable of preventing large or early releases, including containment by-pass; <u>SAM measures to minimize the release of radionuclides into the environment are working.</u> - control locations remain habitable to allow performance of required staff actions.	The extension of the list regarding the assessment of severe accident management measures has been done.	X	Change first proposal to: *- the plant structures, systems, and components (e.g., the containment design), and procedures are capable of preventing large or early releases, including containment by-pass;* Accept second addition.		

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		- planned severe accident management measures are effective.					
Observer WNA 7	7.58	7.58. The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design and not only by implementation of accident management guidelines.	Accident management guidelines are important part of DEC-B management	X	*Covered by Japan 27*		
Japan-27	7.58	The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design ~~and not by~~ combined with implementation of accident management guidelines.	In case of severe accident, flexible measures which combine design and AMG including using mobile equipment should not be excluded.	X	*Change makes it consistent with SSR-2/1 § 5.75 (f).*		
Observer ENISS-68	7.58	The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design and not only by implementation of accident management guidelines.	As it is, it may be understood that SA management should be automatic and should not rely on operator actions.	X	*Covered by Japan 27*		
Germany 38	7.58	7.58. The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design and not by implementation of accident management guidelines.	It is not clear why mitigative severe accident management measures are excluded. One of the main objectives of deterministic severe accident analyses is also to show the effectiveness of SAM measures. Furthermore, for the usage of SAMGs	X	*Covered by Japan 27*		

Reviewer: Country/Organization:		COMMENTS BY REVIEWER Page.... of.... Date: 25/05/16		RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			computational aids are necessary which are developed by deterministic event analyses. Another demand is the ALARA principle mentioned e. g. in 2.18 and 4.6. The compliance of the principle has to be shown also for design extension conditions by deterministic event analyses.				
Canada 63	7.58	Suggest the following changes, The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design and ~~not~~ by operator action credit consistent with the implementation of accident management guidelines	Para 7.65 notes that operator actions should be considered. The implementation of accident management guidelines is consistent with credits for operator action.		*Covered by Japan 27*		
Observer EC/JRC 76	7.58/All	*Please see rationale*	The entire para should be clarified, in particular providing the rationale for not crediting for actions included in the accident management guidelines, at the same time clarifying what is intended to mean by 'design' in apparent opposition to accident		*Covered by Japan 27*		

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			management: fundamental provisions incorporated through backfitting will obviously be reflected in accident management, to an extent that performing actions of mitigating systems used in DEC-B like events will likely be restricted to accident management guidelines, e.g. containment flooding to mitigate MCCI. In addition, take also para 7.65 into consideration for potential updating.				
Germany 39	7.60	7.60. Technical acceptance criteria should ensure that containment integrity is maintained. Examples of acceptance criteria for design extension conditions analysis would include limitation of the containment pressure, temperature and hydrogen concentration and stabilization of molten corium.	That paragraph is incomplete because another upstream safety goal is the prevention of RPV failure. For this, acceptance criteria can also be listed, like retention of core melt inside RPV, external cooling of RPV etc..	X	*Change to:* 7.60. Technical acceptance criteria should ensure … extension conditions analysis ~~would~~ could include limitation of the containment pressure, maintaining in-vessel retention, temperature and hydrogen		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					concentration …"		
Observer EC/JRC 77	7.60/3	… temperature and ~~hydrogen~~ flammable gases concentration and stabilization of molten corium.	Though highly plant-dependent (in particular basemat chemical composition dependent), long-term combustion process are more governed by carbon monoxide rather than hydrogen generation. Therefore, it is recommended to replace hydrogen by flammable gases throughout the text.	X			
Observer EC/JRC 78	7.62/(Addition)	*Please see rationale*	Application of para 7.33, bullet 3, is much more related to severe accidents than DBAs. Therefore it should be added here as well –even if mentioned within the 'available systems' subsection. For instance, ongoing IAEA-TECDOC-1135 on "ASSESSMENT OF NUCLEAR POWER PLANT EQUIPMENT RELIABILITY PERFORMANCE FOR SEVERE ACCIDENT CONDITIONS" led by A. Duchac from IAEA focuses exactly on this topic and ways to tackle with it. The			X	*Bullet 2 of existing text covers survivability of equipment.*

		COMMENTS BY REVIEWER		RESOLUTION			
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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			author is invited to look it up.				
Germany 40	7.64	7.64. Single failure criterion need not be considered in severe accident analysis. Furthermore, no additional failure of a system/component due to maintenance has to be considered.	See Comment 33.	X	*Covered by Canada 30*		
Canada 30	7.64	7.64. Single failure criterion need not be considered in severe accident analysis and unavailability due to maintenance does not need to be considered.	Make it clear that the requirement on safety systems in para 7.36 of the guide does not apply in DEC.	X			
Czech 22	7.65.	Operator actions should be considered as for design extension conditions ~~without~~ to mitigate significant fuel degradation.	These are core melting sequences and melting is significant fuel degradation.	X	Change to: "7.65. The same operator actions should be considered as for design extension conditions without significant fuel degradation. See paragraph 7.52."		
France 20	7.65 (new)	Non-permanent systems and equipment should not be considered for demonstration of adequacy of the nuclear power plant design in the short term phase of an accident. Such equipment is typically considered to operate for long-term sequence and is considered available in the	This applies also for severe accident.		*New paragraph:* 7.64A. Non-permanent systems should not be considered for demonstration of adequacy of the		

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		development of emergency operating procedures or accident management guidelines			NPP in the short term. Such equipment is typically considered to operate for long-term sequence and is considered available in accordance with emergency operating procedures or accident management guidelines. The time claimed for availability of non-permanent equipment should be justified; for example, for new NPPs non-permanent equipment may be credited after 8 hours for equipment stored on site or 72 hours for equipment stored off site.		
Czech 23	7.66	Release and transport of fission products, including filtered venting to prevent overpressure in the containment;	Venting thorough sand bed filters or scrubbers, etc. Not direct venting to atmosphere. Scrubbers are one of the primary devices			X	*This is a list of phenomena, not a set of design requirements. The efficiency of filters (if*

Reviewer: Country/Organization:		COMMENTS BY REVIEWER Page.... of.... Date: 25/05/16		RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			that control gaseous emissions in case of emergency.				*any) should be modelled.*
Observer EC/JRC 79	7.66/7	In-vessel melt retention by RCS injection at different degrees of core damage, and by ex-vessel cooling	Clarification's sake: it is not very clear what 'in-vessel retention' is meant to be.			X	*In-vessel retention can be quite different in different designs, e.g. PHWR. Current text is sufficiently general to cover this.*
Observer EC/JRC 80	7.66/8 (addition)	Direct Containment Heating	Even if the list is not exhaustive, DCH is comparable to steam explosions and combustion processes so it should be included for clarification's sake.	X	Add new bullet.		
Japan-28	7.68	According to ~~Requirements to be met include~~ Req. 20 from SSR-2/1 (Rev. 1), § 5.31 [1], "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'."	Clarification for "practically eliminated".			X	*Not to include quoted text from requirements*

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Observer EC/JRC 81	7.69/1	According to ~~2.8~~ 2.1,…	Typo	X			
Observer ENISS-69	7.70	Ask for clarification.	Consistency between 2nd and 3rd bullets should be improved. 2nd bullet requires a high confidence demonstration. Then, 3rd bullet requires sensitivity studies. Shouldn't these sensitivity studies be part of the high confidence demonstration?		*See proposal from France*		*Text of bullets 2 and 3 combined and clarified.*
Canada 31	7.70	7.70. Demonstration of practical elimination of certain conditions (unless such conditions are judged as physically impossible) should include, where appropriate, the following steps: […]	Deterministic safety analysis is not always needed (last two bullets). For example, catastrophic pressure vessel failure is not analysed.	X			
Observer EC/JRC 82	7.70/7	Sensitivity studies ~~to provide assurance that sufficient margins exist to address uncertainties and~~ to avoid cliff-edge effects	It is the opinion of this reviewer to make a clear distinction between uncertainty and sensitivity as they constitute very different		See changes to ENISS comment on 7.70.		

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			statistical tools even if sharing some of their tasks. Uncertainty margins cannot be assessed through sensitivity analysis as para 7.70 is suggesting. Moreover, such complex, interrelated uncertainties, as those characterizing the field of severe accidents, would need to be integrally taken while sensitivity analysis is usually performed on one-at-a-time basis. Instead, cliff-edge effects can be deterministically imposed by forcing the code to simulate the worst conditions and afterwards then check whether outcomes go beyond design limits.				
Observer WNA 8	7.70 bullet 2	Assessment of the ability of the design and operational provisions with high confidence to eliminate or to address the challenges, <u>by providing an appropriate combination of safety classified features</u>	Practical elimination cannot be based on non classified features	X	*Covered by changed proposal from ENISS.*		

Reviewer: Country/Organization:		COMMENTS BY REVIEWER Page.... of.... Date: 25/05/16		RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
France 8 France-21?	7.70	Demonstration of practical elimination of certain conditions (~~unless such conditions are judged as physically impossible~~) should include the following steps: • Identification of undesired conditions (challenges) potentially endangering the containment integrity or by-passing the containment, resulting in early or large releases, • Challenges should be addressed. In case this is not possible, design and operational provisions should be implemented in order to practically eliminate them ~~Assessment of the ability of the design and operational provisions with high confidence to eliminate or to address the challenges~~ • Sensitivity studies to provide assurance that sufficient margins exist to address uncertainties and to avoid cliff-edge effects • Final confirmation of the adequacy of the provisions by deterministic safety analysis, complemented by probabilistic safety assessment and engineering judgment.	This § is not clear : - Physical impossibility could be a way for practical elimination, - What does "eliminate with high confidence" means - second bullet is understood according to the proposal - For step 3: during the identification phase, the "threshold values" which should not be exceeded to avoid cliff-edge effects are identified (e.g. the value of the reactivity insertion which can lead to prompt criticality). Once these values are determined, the reactor is designed such to guarantee those margins. Here the object of the sensitivity studies is not clear. **Clarification is necessary (proposed modification is a minimum) or consider full deletion**	X			
Observer ENISS-70	7.71	Although probabilistic targets can be set, demonstration of practical	The practical elimination is relevant for early and large	X	*Changed to:* 7.71 Although		

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Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		elimination of early and large releases should not be based solely on low probability numbers. The achievement of any probabilistic value cannot be considered as justification for not implementing ~~reasonable design or operational measures~~ reasonably practicable safety improvements.	releases and should be named like this. Second addition needed to be in line with SSR 2-1 (especially para 1.3).		probabilistic targets can be set, demonstration of practical elimination of event sequences that would lead to an early radioactive release or a large radioactive release should not be based solely on low probability numbers. The achievement of any probabilistic value cannot be considered as justification for not providing ~~reasonable design or operational measures~~ reasonably practicable safety features.		
Belgium 6	7.72	“Where a claim is made that ~~is~~ the conditions potentially resulting in early or large releases are ‘physically impossible’, …”	Typographical correction (delete “is”)	X			

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Observer EC/JRC 83	7.72/2	… it is necessary ~~to examine the inherent safety characteristics of the system~~ to demonstrate that the conditions cannot~~, by the laws of nature,~~ take place whether because of laws of nature (physically impossible to occur) or because of relying on systems whose inherent fully –or almost fully– passive nature leads to highly confident levels of performance.	The 'practically eliminated' condition is defined in the overarching Safety Requirements document, SSR-2/1, Rev. 1, where it is mentioned that "physical impossibility of the phenomenon with a high level of confidence to be extremely unlikely to arise". Para 7.71, and even more 7.72 when talking about "inherent safety characteristics… by the laws of nature", seems to go too far because of not attending the definiens clause on 'extremely unlikely' hence accounting for risk hence for probability to occur; and because of not considering that in most cases the pursued elimination is achieved through mitigating systems, and even if these systems are passive, they can fail. In fact, passive safety systems belonging to 3rd generation have an associated probability of failure (huge literature is found on that). For instance, if overpressurization as the cause for containment failure is said to be avoided by means of FCV, even if this system were fully			X	*Comment is not clear.*

COMMENTS BY REVIEWER Reviewer: Page.... of.... Country/Organization: Date: 25/05/16				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Observer EC/JRC 84	7.72/10	… elimination by **physical** impossibility of the conditions).	Please see rationale of previous comment 83	X			
Observer EC/JRC 85	7.72	An example dealing with high-level performance could be the Passive Autocatalytic Recombiners to avoid reaching DDT conditions jeopardizing containment integrity. Due to their passive nature, failing to succeed in accomplishing with their committed safety function turns to be extremely unlikely.	Please see rationale of previous comment 83			X	PARS can be impaired due to surface contamination or may have insufficient surface area to deal with the threat.

Sections 8
DS491 Draft Safety Guide: Deterministic SA for NPPs - Step 7

COMMENTS BY REVIEWER Reviewer: Page: Country/Organization: Date:				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
USA 8	8.3 Line 2 (p. 48)	Safety report should provide a list of all plant states considered in the deterministic safety analysis, appropriately grouped according to their frequencies and specific challenges to the integrity of physical barriers against releases of radioactive substances.	Editorial	X			
FIN-5	8.5 Lines 3-4	Brief description of the computer codes used in the deterministic safety analysis should be provided. In addition to the reference to the specific code documentation the description should contain convincing justification that the code is adequate for the given purpose and has been *verified and* validated by the user as described *in para. 5.13 – 5.36.*	Clarification: Verification and validation both should be considered. The reference to relevant paragraphs in the document would be good.	X	"… given purpose and has been verified and validated by the user (see §5.13 to §5.36) ~~to a reasonable extent~~."		
CAN 32	8.5 Lines 3-4	8.5. Brief description of the computer codes used in the deterministic safety analysis should be provided. In addition to the reference to the specific code documentation the description should contain convincing justification that the code is adequate for the given purpose and has been validated by the	Suggest "validated to an appropriate extent". Reasonable confidence is OK for DEC. We want high confidence for DBA. The different requirements for each plant state is captured in para 8.7 below.	X	*See resolution to FIN-5. It covers the proposal*		

		user to ~~a reasonable~~ an appropriate extent.					
GER 42	8.7 Line 2	8.7. The simulation models and the main assumptions used in the analysis for demonstrating compliance with each specific acceptance criterion should be described in detail ~~introduced~~, including description of the scope of validation of the model. This description should include potentially different approaches used for each plant state.	It is very important especially for the review of the computational results to describe the input deck of the plant under examination and the assumptions made in detail. Otherwise, for the reviewer it could be hard to understand the results of the analyses.	X	"… compliance with each specific acceptance criterion should be described in detail ~~introduced~~, including ~~description of~~ the scope of validation …"		
CAN 33	8.9 Line 2 and last line	8.9. The time span of any scenario analysed and presented should extend up to the moment when the plant reaches a safe and stable end state (not all sensitivity calculations need to be presented over the full time scale). What is meant by a safe and stable end state should be defined. Typically it is assumed that a safe and stable end state is achieved when the core is covered and long term heat removal from the core and/or containment is achieved, and the core is subcritical by a given margin.	Sensitivity calculations are not normally presented over the full time scale. Also, for many scenarios, heat must be removed from containment as well as the core.	X	*First comment:* "…reaches a safe and stable end state (typically not all sensitivity calculations need to be presented over the full time scale). What is meant…" *Second:* ":… and long term heat removal from both the core and the containment is achieved, and the core is …"		
CAN 64	8.9 Line 3	Suggest the following changes, Typically, it is assumed that a safe and stable end state is achieved when the core is covered and long term heat removal from the core is achieved, (established controlled venting from	For a multi-unit PHWR with negative pressure containment, a safe and stable end state may include controlled venting from the containment		*See resolution to CAN-33. It is somehow covered there*		*The wording suggested seem too detailed; better to use the one from CAN-33*

		the negative pressure containment in a multi-unit PHWR) and the core is subcritical by a given margin.					
Observer ENISS-71	8.9	Move to 7.27	Inconsistent here. Rather in 7.27.			X	*The adequacy of the location has been checked and confirmed. Chapter 7.27 seems not to be the adequate place.*
CAN 34	8.16 Line 2	8.16. In case of the need, the safety analysis should be reassessed to ensure that it remains valid and meets the objectives set for the analysis. The results ~~shall~~ should be assessed against the current requirements relevant for deterministic safety analysis, applicable experimental data, expert judgment, and comparison with similar analyses.	This is a guidance document. Change to "should" or refer to the standard this requirement is taken from.	X	Editorial		
ALGE 1	8.16 Line 2	The results should be assessed against the current requirements relevant for deterministic safety analysis, applicable experimental data, expert judgment, and comparison with similar analyses.	DS491 is drafted as a safety guide.	X	Editorial		
CAN 65	8.17 Line 1	Suggest the following changes, *8.17. The outcomes of the reassessment including new deterministic safety analyses if necessary should be reflected in updated the safety report with ~~the same~~ an appropriate level of comprehensiveness ~~as the original safety report~~ commensurate with the extent of changes being considered and the potential impacts.*	The level of comprehensiveness of new deterministic analysis should be commensurate with the extent of changes and their impacts being assessed.	X	*Formulation: "… analyses if necessary should be reflected in the updated ~~in the~~ safety analysis report with ~~the same~~ an appropriate level of comprehensiveness ~~as the original safety report~~*		

					commensurate with the extent of changes and the associated impacts."		
USA 9	8.17 Line 2 (p. 49)	The outcomes of the reassessment including new deterministic safety analyses if necessary should be reflected in the updated ~~the~~ safety report with the same level of comprehensiveness as the original safety report.	Editorial	X	*Covered with the resolution provided to CAN-65*		

Section 9 and Annex
DS491 Draft Safety Guide: Deterministic SA for NPPs - Step 7d

COMMENTS BY REVIEWER Reviewer: Page: Country/Organization: Date:				RESOLUTION			
Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Observer ENISS-72	9.1 line 1	Ask for clarification	It should be explained that the word "verification" is used here for "surveillance" of work performed by other entities. It is not used for Verification as performed in V&V.			X	*The clarification seems not necessary; see GSR Part 4 Req. 21*
KOR-7	§9.2 Line 3	"…to reconfirm that the safety analysis developed by other entities such as designers, manufacturers and constructors has been carried out in an acceptable way and satisfies the applicable safety requirements.	Rewrite the sentence based on the para 4.67 in GSR Part 4)	X	*Consistency with GSR Part 4, para 4.67*		
CAN 35	9.14 Line 2	9.14. All numerical models used in safety analysis should show their reliability through comparisons, independent analyses and qualification, with the aim of ~~guaranteeing~~ demonstrating that their intrinsic uncertainty level complies with the reliability required for the whole design project.	"Guaranteeing" is OK for DBA, but too strong for DEC analysis. Suggest "demonstrating".	X	Clarification		
CAN 66	9.15	Suggest the following additional bullets,	The components of independent verification should include selection of	X	"… • Selection of acceptance		

		Selection of safety analysis method *Selection of safety analysis computer codes and adequacy of code validation*	safety analysis method and computer codes & adequacy of validation		criteria • Selection of safety analysis method • Selection of safety analysis computer codes and adequacy of code validation • Selection of assumptions for ensuring safety margins • …		
Observer ENISS-73	9.15	After "in accordance … independent calculations", please add: "*The independent verification should be fit to purpose and, depending of the safety analysis, should determine which of the three following verification levels is the most adequate:* *Level 1: compliance with the specifications of the study* (introduce here the bullet points of the paragraph) *Level 2: level 1 + critical analysis of the assumptions of the study and verification of the orders of magnitude of the results* *Level 3: level 2 + independent calculations*"	An independent review is seen beneficial, but an independent calculation of certain values might be useful and proportionate only in certain cases. Section 9 as a whole does not explain it sufficiently. See GSR part 4 (Rev. 1) §4.69 which says: "specific review MAY contain comparison … with independent calculations". The proposed new text says how the licensee can take it into account			X	Outside the scope of this Safety Guide (too detailed)
Observer ENISS-74	9.16		Clarify the meaning of "if code models were developed independently"	X	9.16. An independent check of selected computer … can meet the objectives		

					of the review if plant ~~code~~ models (including nodalization, initial and boundary conditions) were developed independently		
Observer WNA 9	9.17	9.17. If independent calculations are performed, it may be appropriate	Performing independent calculations is not a requirement, it should not be considered as systematic	X	"9.17. ~~Regarding selection of cases for~~ If independent calculations are performed, it may be appropriate to select ...."		
ANNEX							
Observer EC JRC-86	Annex A.1 (addition)	(j) Design specifications, e.g. sizing, capacity, setpoints, environmental bounding conditions for equipment qualification, etc., for existing and new, backfitted mitigating systems.	This application concerning severe-accident simulation codes is crucial. As it is related with backfitting, it does not fall under A.1(a) category. Para A.2 should be updated correspondingly.			X	*Part of A1 (a) [and (e)]*
USA 10	A.2 (p. 56)	Deterministic safety analysis associated with the design and authorization (licensing) of a nuclear power plant (items (a) to (e)) may be performed to demonstrate compliance with established acceptance criteria with adequate safety margins (ensured in different ways for DBAs and design extension conditions).	Added missing parenthesis.	X	Editorial		
CAN 36	A.2. last sentence	Deterministic safety analysis associated with analysis of operational events, development of procedures or	Suggest changing "possible" to "practicable".	X	Clarification		

		guidelines and support of the PSA (items (f) to (i)) are typically not aimed at demonstration of compliance with acceptance criteria and are performed in a realistic way to the extent ~~possible~~ practicable.					
CAN 37	A.5 Line 2 And Last line	A.5. The designer typically uses the safety analysis as an integral part of the design process, which ~~typically~~ normally consists of several iterations which may continue through the manufacture and construction of the plant. The safety analysis used in the design is performed according to a quality assurance (QA) programme ~~which includes independent reviews of all design documents~~.	Suggest changing "second occurrence of "typically" to avoid repetition. The final clause does not seem to relate to DSA. However, if it is retained, change to "key design documents". I suspect that an independent review of all design documents is not done.	X	*Second change is accepted* (Editorial / Clarification)	X	*First change: Typically is used in the SG*
CAN 38	A.17	Format the list of objectives as a numbered list.	Format the list of objectives as a numbered list.	X	Editorial *(Bulleting (a) … (i) will be included. They were lost in formatting)*		
CAN 39	A.20 First sentence, Line 4	A.20. Best estimate deterministic safety analyses are typically performed to confirm the recovery strategies that have been developed to restore normal operational conditions at the plant following transients due to anticipated operational occurrences and DBAs and design extension conditions without ~~core melt~~ significant fuel degradation. […]	Change "DEC without core melt" to "DEC without significant fuel degradation".	X	Editorial		
CAN 40	A.25 Line 4 and Line 10	Delete "light". Two occurrences.	Para A.25 would apply to heavy water cooled reactors too. Suggest	X	Editorial *(Used*: ~~light~~ water cooled reactors")		

			deletion of “light”.				
CAN 41	A.29 Line 3	A.29. More specifically, the deterministic analysis is performed to specify the order of actions for both automatic systems as well as operator actions. This determines the time available for operator actions in specific scenarios, and ~~to specify the~~ supports the specification of success criteria for required systems for prevention and mitigation measures.	Suggest “and supports specification of the success criteria”	X	Clarification *Additionally, the following sentence will be added at the end of A.27:* “…However, it is acknowledged that some residual risks will remain.”		