Review of Safety Guide on Deterministic Safety Analysis for NPPs (DS491)

Deadline for comments by Member States: 7 December, 2016

Table of Member States comments Resolutions

Comments provided by Member States (invited reviewers):
On schedule: Tajikistan (no comments); Pakistan; Armenia; Switzerland; Finland; Turkey; Spain; Sweden (no comments); France; Czech Republic; Japan; Canada and India // After deadline: UK (13 Dec); USA (14 Dec); Iran (21 Dec); Russia (22 Dec) and Jordan (18 Jan, 2017).

Other comments provided (unsolicited):
On schedule: EC-JRC; ENISS and IAEA/NSRW/WES // After deadline: ILO -International Labour Organization- (13 Jan, 2017).

6 April 2017

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
General comments							
Finland 1	General	The update of the Safety guide Deterministic Analysis for Nuclear Power Plants has been considered one of the IAEA Safety Standards needing urgent updating. It is good that that the safety guide is available for MS comments with recently updated IAEA Requirements.		X	*(Comment noted and appreciated)*		
ENISS - 1 (Observer/ unsolicited)	General comment	In the whole guide : - Replace "probabilistic safety analysis" by "probabilistic safety assessment"	It is better to use the term defined in the 2016 IAEA Safety Glossary.		*Both terms are used in the IAEA Safety Glossary 2016 and in GSR Part 4 (Rev.1). The use of these terms has been reviewed and paras 3.18 and A-1 (i) have been updated*		

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IAEA/WES-44 (Unsolicited)	General	Note: The abbreviation 'SSCs' is used in Paras 3.54 and 4.17 of this Safety Guide, but nowhere explained in the text. The abbreviation should be introduced in the 1st sentence of Para 2.1 and then consistently used throughout the entire document – namely in Paras 2.5 (a), 3.14, 3.50, 7.56 and A-13.	Editorial.		*The abbreviation SSC used in paras 3.54 and 4.17 has been replaced by the terms in full*		
IAEA/WES-45 (Unsolicited)	General	Note: According to SPESS C, Annexes are not an integral part of an IAEA Safety Standard. All references made in the paragraphs of the Annex to DS491 (Application of DSA) have to be listed separately at the end of the Annex.	To be in line with SPESS C (Guidelines for drafting IAEA Safety Standards and Nuclear Security Series publications), Version 3.1 dated 16 November 2015.	X			
Turkey 6	General	Guidance provided in this draft Safety Guide is not consistent with 5.26 of SSR-2/1 (Rev.1). Even if this draft Safety Guide provides a guidance for the Safety Report, SSR-2/1 (Rev.1); it is proposed to modify the expression in 5.26 of SSR-2/1 (Rev.1) as "The design basis accidents shall be analysed in a conservative manner. This approach involves postulating certain failures in safety systems, specifying design criteria and using ~~conservative assumptions, models and input parameters~~ conservative assumptions with conservative and/or	As SSR-2/1 (rev.1) From the statement stated in 5.26 of SSR-2/1 (Rev.1), it is understood that as well as the assumed plant conditions, both the physical models used and the input and boundary conditions used shall be set <u>conservatively</u>; i.e., only Option 1 defined in DS 491 can be used according to SSR-2/1 (Rev.1). However, in DS 491, three different conservative			X	SSR-2/1 (Rev. 1) is already published and the request cannot be taken into account currently. (It can be kept in mind for future reviews). Regarding Ds491, Best-estimate models with appropriate allowance for uncertainties can still deliver an overall conservative result; hence DS491 is consistent with the intent of SSR-2/1 (Rev.

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		best estimate models, and input parameters in the analysis." And eventually to modify 6.4 of DS491 in accordance with the modification in 5.26 of SSR-2/1 (Rev.1).	analysis defined that "assumptions which are made about the availability of plant systems" are conservative but models and input parameters can be conservative and/or best estimate. This incostistency should be eliminated.				1)
Turkey 7	General	It is suggested to mention more explicitly that safety classification of SCCs is associated with deterministic safety analysis by highlighting the interface with the IAEA document **SSG-30**.	The document does not present any reference to show the relevance of deterministic safety analysis in safety classification.		*At the end of para. 7.2 it will be added:* "… on in this section. DSA should only credit SSCs that meet the requirements associated with relevant plant states, with due consideration of safety classification (see SSG-30) [20]."		
Turkey 8	General	It is proposed to mention related reports among **IAEA Safety Reports Series** which provide information on how to perform deterministic safety analyses such as SRS-23, SRS-30, and SRS-56.	The IAEA documents associated with the deterministic safety analysis should be mentioned and referred to provide guidance on how to reach the existing detailed information on DSA.			X	Taking into account the hierarchy of IAEA documents higher level should not refer to lower level documents. Also, the IAEA Safety Report Series indicated in the comment, are obsolete, do not take into account applicable requirements, focus DBA analysis and should no longer be

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							relied upon.
Turkey 9	General	Evaluation of the event sequences and source terms that correspond to different emergency classes is suggested to be referred among the usage areas of deterministic analysis.	Explanation regarding determination of the EALs which is the backbone of the triggering mechanism for initiation of emergency response is only given in GSG 2 among the recent documents of the IAEA. Furthermore, the link between the EALs and deterministic safety analyses is not provided. However, the EALs should be determined in the licensing stage before commissioning of the NPPs. Thus the importance of deterministic safety analysis in terms of determining EALs should be emphasized in this document. The details can be given in another document which is related to emergency preparedness and response.			X	Outside the scope of DS491. This Safety Guide covers the calculation of source term in the different plant states but not emergency preparedness and response. *(Note from TO: The text indicated under "Reason" in this comment is repeated in Turkey 12 (para. 2.18). Also in Turkey 17, although this comment is not included in the Table; no para. or comment is provided there, only the "Reason")*
Iran 1		A separate annex is recommended to be added to include typical path and steps that should be taken by the licensee of the purpose of independent verification of deterministic safety analysis calculations	What is brought under item 9.16-9.18 of DS491 , covers general aspects of verification without detailed explanation on performing different tasks.			X	Guidance provided covers key aspects to be taken into account. The level of detail requested is even not provided on how to perform

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			A separate annex could include more guidelines, recommended good practices and relevant reference on the subject.				deterministic safety analysis, so it is not provided for its independent verification. That level of details is defined by each State.
Iran 2		More elaboration on the proper methodology of regulatory body for independent verification of deterministic safety analysis should be made	Under A-11 of DS491 a short statement if made on this subject. More explanations and recommendations could be included as to the approach and details of the process of verification by Regulatory Body and its relevance to the activity of the licensee.			X	See Resolution to "Iran-1". Each State defines these practices.
Section 1							
Russia 3 comment 1	1.1	This Safety Guide provides recommendations and guidance on use of deterministic safety analysis and its application to power generating Unit of NPP rather than NPP as a whole.	Point 1.11. This Safety Guide focuses on neutronic, thermal-hydraulic, fuel (fuel channel for pressurized heavy water reactor) and radiological analysis. However: The analysis of system reliability is not mentioned and not considered in this Guide.			X	Paragraph 1.1 is necessary and seems correct. Further details regarding the scope of the Safety Guide are provided in para. 1.11
Russia 2	1.2,	Deterministic safety analyses for	To supplement the list of			X	The change seems not

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Comment 1	Second sentence	normal operation, anticipated operational occurrences, design basis accidents and design extension conditions including severe accidents and justification of the operator's actions in course of scenarios listed above, as defined in Ref. [1] and in the IAEA Safety Glossary [3], are essential instruments for confirming the adequacy of safety provisions.	tasks of the deterministic analysis given in this sentence with a task to justify the operator's actions during the above listed events, where such actions are specified in the design.				necessary. Paragraph 1.2 provides background and operator's actions are covered by the existing sentence. In addition, justification of operator's actions is not part of the deterministic safety analysis, which just considers those actions to see whether the criteria are met. (See para. A-20)
Turkey 1	1.3 Line 6	"... Updating of the Safety Guide is also aimed at ensuring consistency with current IAEA Safety Standards, including updating of Safety Standards implemented with lessons learned from the Fukushima Dai-Ichi nuclear power plant accident."	Lessons are learned from the accident, not from the NPP.	X	*Para 1.3 will be modified as follows:* "… with lessons ~~learned~~ from the Fukushima Daiichi nuclear power plant accident."		
India	1.3 Last sentence	**"…**Updating of the Safety Guide is also aimed at ensuring consistency with current IAEA Safety Standards, including updating of Safety Standards implemented with lessons learned from the Fukushima Daiichi nuclear power plant accident."	Lesson are learnt from the 'Accident' and not from the normal operation of the Fukushima Daiichi nuclear power plant	X	*See Turkey 1*		
Canada 1	New para after 1.3.	***This is a major comment.*** 1.3A. It might not be practicable to apply all the ~~requirements~~ **guidance** of this Safety ~~Requirements~~ **Guide** publication to nuclear power plants	This safety guide lacks a paragraph equivalent to SSR-2/1, paragraph 1.3 that describes how it is applied to plants built to		*Last sentence of para 1.6 will be modified as follows:* "… The guidance provided is intended		

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		that are already in operation or under construction. In addition, it might not be feasible to modify designs that have already been approved by regulatory bodies. For the safety analysis of such designs, it is expected that a comparison will be made with the current standards, for example as part of the periodic safety review for the plant, to determine whether the safe operation of the plant could be further enhanced by means of reasonably practicable safety improvements.	earlier standards. This is particularly problematic for the numerical criteria (for example meeting the operator response times given in paragraph 7.37). It is strongly suggested that such a paragraph is added. The proposed text shows the changes needed to the SSR-2/1 para 1.3 to make it applicable to a Safety Guide.		to be as much as possible consistent with paras 1.3 and ~~§~~1.6 of SSR-2/1 (Rev. 1) [1] and it is particularly based on experience with deterministic safety analysis for water cooled reactors."		
UK 1	1.5 Line 4 (Last sentence)	*Replace:* "…for all plant states. ~~Deterministic safety analyses are required to determine the characteristics of the releases (source term) depending on the status of the barriers for different plant states~~." *With:* "…for all plant states. Deterministic analysis is carried out primarily to determine the design parameters for safety functions; to ensure that barriers to release of activity are preserved as far as reasonably practicable. Analysis is also used to inform the selection of mitigation measures, should protection fail. This is done by determining the potential effect on the public and	The analysis is not simply intended to quantify releases. The general expectation is that these should be prevented – we design for success. The most important analysis is design-basis analysis which is used to set protection parameters.			X	Previous sentence says: "…analyses that are required to demonstrate adequate fulfilment of safety functions in order to ensure that barriers to the release of radioactive material will prevent an uncontrolled release…". So the additional sentences requested seem not necessary.

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		operating staff of a postulated event scenario.”					
EC-JRC 1 (Observer/ unsolicited)	1.5 Line 6	“… depending on the status of the barriers for different plant states, and to demonstrate compliance with acceptance criteria”.	Deterministic safety analyses do not only aim at characterizing the source term (e.g. LOCA simulations for containment P-T characterization).			X	The request seems covered with the previous sentence. Compliance with acceptance criteria is stated in para. 1.9. (See also resolution to comment UK 1).
UK 2	1.6 Line 2	“…power plant and, ~~as far as reasonably practical or achievable,~~ also the safety…”	The clause relates to the analysis not the measures and this caveat is not relevant or not clear: The scope of analysis required should not generally be affected by the plant status (although the practicable mitigation measures may be).			X	The assumptions that are used (or can be used) in analyses depend on the plant status (designed according to new requirements or designed and constructed 30 years ago with old requirements). See also Canada 1.
Russia 3 Comment 2	1.6.	Do indicate possibility to use recommendation of this Guide so: provided in examples list of accidents oriented on new NPPs may be brought up for previous generation NPPs if listed accident will be applied to more heavy category	1.6 This Safety Guide focuses primarily on the deterministic safety analysis for the design safety of new 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			X	It seems not necessary to add this type of list of examples, which could also limit the scope and lifetime of the Safety Guide.

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			review their safety assessment. *However: provided in examples list of accidents is oriented on new NPPs, using passive safety systems. For previous generation NPPs brought accident pertain to more heavy category.*				
Russia 2 Comment 2	1.7	1.7. The guidance provided in this Safety Guide focuses on best practices in the analysis of all plant states considered in the design, from normal operation through anticipated operational occurrences and design basis accidents up to design extension conditions including severe accidents. Footnote: The concept of "design extension conditions" established in IAEA Safety Standard SSR-2/1 is not accepted in all countries. For example, in Russia the former concept of "beyond design basis accidents" is in use, which means that a representative set of all possible accident scenarios is analyzed irrespective of their probability.	The concept of "design extension conditions" established in the IAEA standard SSR-2/1 is not accepted in all countries. So, for example, in Russia this concept is not accepted, and the former concept of "beyond design basis accidents" is still being used. At this, in the design the representative set of all possible events irrespective of their probability is considered. Scenarios which probability is higher than the goal reference points established in federal norms and rules (OPB NPP) are considered for the purpose of developing the supplementary			X	This Safety Guide does not describe whether or how the States apply the safety requirements from SSR-2/1 (Rev.1) or other requirements. It provides guidance on how the requirements of SSR-2/1 (Rev.1) can be fulfilled. "Design extension conditions" is the term currently used in IAEA Safety Standards.

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			technical solutions, including special technical means on beyond design basis accidents management, and for scenarios with probability smaller than the established goal reference points – with the purpose to take account when developing a beyond design basis accidents management guidance. For both these purposes the corresponding deterministic analyses are required. It should be reflected in a footnote to this para.				
EC-JRC - 2 (Observer / unsolicited)	1.8 Line 1	"…reactor coolant system, containment, fuel storage…"	DEC figure of merit is not anymore the core or vessel but the containment	X	*Paragraph 1.8 will be modified:* 1.8. Regarding deviations from normal operation ~~T~~this Safety Guide deals with human errors and ~~those~~ failures of plant systems (e.g. systems from ~~in~~ the reactor core, reactor coolant system, containment, fuel storage or~~,~~ other systems containing		

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					radioactive material~~s)~~ ~~or any other system that has~~ having the potential to affect…"		
IAEA/ WES-1 (Unsolicited)	1.8 1sentence	"1.8. This Safety Guide deals with those failures in the reactor core, reactor coolant system, fuel storage, systems containing radioactive materials or any other system that ~~has~~ have the potential to affect the performance of safety functions …"	Grammar.	X	*See EC-JRC-2*		
Turkey 18	1.9 Lines 3-4	Deterministic safety analyses use computer codes which, as analytical tools, have broader range of applications.	Original sentence seems out of context.		*Last sentence will be deleted*		
Canada 2	1.11 Line 3	1.11. This Safety Guide focuses on neutronic, thermal-hydraulic, fuel (fuel channel for pressurized heavy water reactors) and radiological analysis.	Editorial. Heavy water reactors should be plural.	X			
IAEA/WES-2 (Unsolicited)	1.12 1 sentence	"The extent of radiological analysis in this Safety Guide includes the transport of radioactive substances within the buildings and structures of the nuclear power plant, …"	Grammar.	X			
Turkey 19	1.12 Line 9	"…minimization of radiation sources, appropriate nuclear power plant configuration, adequate shielding…"	Each measure to be taken needs and has an adjective, except "shielding and ventilation design".	X			
India 2	1.12 Last sentence	"…Determination of the doses to personnel at the nuclear power plant ~~Staff~~ is therefore not covered by this Safety Guide."	The word 'staff needs to be deleted as already 'personnel' is used in the sentence	X			

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Turkey 20	1.13 Line 1	1.13. This Safety Guide also covers aspects of the analysis of radiological releases of radioactive materials/substances, up to and including the…"	In the Safety Glossary, the term used is not "radiological release" but "release of radioactive materials" or "release of radioactive substances". Also, when using the expression "up to", it is better to emphasize inclusion with "up to and including".		*Paragraph 1.13 will be modified as follows:* 1.13. This Safety Guide also covers aspects of the analysis of ~~radiological~~ releases of radioactive material, up to and including the determination …"		
IAEA/WES-3 (Unsolicited)	1.13 Line 7	"…While general rules for deterministic safety analysis apply also to the analysis of radiological consequences of anticipated operational occurrences and accident conditions, …"	Editorial.	X			
Turkey 21	1.13 Line 7	"…general rules for deterministic safety analysis apply also to analysis of radiological consequences of (such as doses received during)**…"**	Inclusion of an example of actual calculations help clarify the meaning.			X	The change seems not necessary
India 3	1.12 & 1.13	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 net covered by this Safety Guide **(both the paras are conflicting with each other in terms of coverage of source term release to the environment)**	As per SSR-2/1 clause 2.9, requirement-5, clause 5.71 and clause 5.75 (d), one need to carry out the analysis for dose assessment. Since this guide covers the licensing aspect, guidance for analysis should be extended till dose assessment or specific reference should be made			X	Apart from the changes indicated in the Resolution to Turkey 20, no other changes seem necessary. This Safety Guide covers determination of source term and other safety guides continue from such determination, as it is clarified at the end of para. 1.13.

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			regarding where and how these aspects are covered. Other aspects such as shielding, ventilation system design etc. can be left.				
Canada 3	1.15 Line 3	"…While in general, documentation and electronic records related to deterministic safety analysis process and outputs provide~~,~~ limited information regarding …"	Editorial. Delete comma after "provide".	X			
Turkey 22	1.15 Line 3	"…provide~~,~~ limited information regarding equipment location and vulnerability and practically…"	Unnecessary comma needs to be deleted.	X			
Canada 4	1.16 Line 3	"…It includes general statements necessary ~~as basis~~ to provide a basis for specific guidance by other sections of this Safety Guide…"	Reword for clarity.		*Paragraph 1.16 will be modified:* "…It includes general statements necessary as basis ~~to provide~~ for the specific guidance provided in the ~~by~~ other sections of this Safety Guide…"		
Russia 2 Comment 3	1.19, last sentence	"…Section 7 provides specific guidance on performing deterministic safety analysis for each individual plant state and on justification of the operator's actions at the analyzed states.	To supplement this sentence with words: "and on justification of the operator's actions at the analyzed states".			X	It seems not necessary to emphasize operator's actions in this Section/para. Justification of operator's actions is not part of deterministic safety analysis.

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Section 2							
France 1	2.1 Line 10	"…that the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release can be considered as ~~are~~ 'practically eliminated'…"	Wording consistency (see 2.18 (b)	X			
Canada 5	2.1 Last line	"… and that the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive release~~s~~ are 'practically eliminated' (see §3.55)."	Editorial. Second occurrence of "radioactive releases" should be singular.	X			
Russia 2 Comment 4	2.1 Two last lines of the <u>first</u> sentence	2.1. The objective of deterministic safety analysis for nuclear power plants is to confirm that safety functions and the needed systems, structures and components, in combination where relevant with operator's actions, are capable and sufficiently effective, with adequate safety margins, to keep the radiological releases from the plant within acceptable limits. The second objective of deterministic safety analysis for nuclear power plants is to determine and justify operator's actions in case of breach of normal operation, including design based	In this para it is established that the objective of the deterministic safety analysis for nuclear power plants is to confirm that functions of safety and necessary systems, structures and components in a combination with actions of the operator are sufficiently capable and effective to keep with the corresponding safety margins the radiological consequences within the acceptable limits. At the same time the actions of the operator mentioned in			X	Copy of Resolution to "Russia 2 Comment 1" (second part of the justification): *"Justification of operator's actions is not part of the deterministic safety analysis, which just considers those actions to see whether the criteria are met. (See para. A-20)."*

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		accidents.	this para also have to be specified and proved by the corresponding analyses. Therefore after the first sentence of this para it is necessary to insert the new sentence establishing the second objective – determination and justification of actions of the operator at violations of normal operation, including accidents				
Turkey 23	2.1 Line 4	"…to keep the ~~radiological~~ releases of radioactive materials/substances from the plant within acceptable limits. Deterministic safety analysis…"	See Comment No: 3 *(Note TO: it seems to refer to Turkey 20)*	X			
Russia 2 Comment 5	2.1, two last lines of the third sentence	"…and that the possibility of certain conditions arising that could lead to an early radioactive release or a large radioactive releases are 'practically eliminated' (see §3.55). Footnote: The concept "practically eliminated", established in IAEA Safety Standard SSR-2/1, is not accepted in all member states. For example, in Russia this concept which was called earlier the concept of "hypothetical accidents" was rejected after Chernobyl accident.	The concept "practically eliminated", established in the IAEA standard SSR-2/1, is not accepted in all countries. So, for example, in Russia this concept which was called earlier the concept of "hypothetical accidents" was rejected after Chernobyl accident. Therefore it is necessary to make here a footnote with the corresponding explanation.			X	1) See Resolution to Russia 2 Comment 2, about para. 1.7: *[This Safety Guide does not describe whether or how the States apply the safety requirements. It provides guidance on how the requirements can be fulfilled.]* 2) Practical elimination is treated in, but not limited to, the IAEA Safety Glossary, SSR-2/1 (Rev.1) and TECDOC 1791

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							("Considerations on the application of the IAEA safety requirements for the design of NPPs")
Russia 3 Comment 3	2.1.	The List of objectives of deterministic safety analysis shell be supplemented with proposition that its results are used for optimization control response under severe accident, providing most "soft" behavior of accidents with standpoint of the restriction of the spreading to radioactivity.				X	Deterministic safety analysis (DSA) is not used for "optimized control response". Paragraph 1.9 indicates the use of this Safety Guide regarding DSA.
EC/JRC-3 (Observer / unsolicited)	2.1 Lines 1 and 4	"…is to confirm that ~~safety functions and~~ the needed systems … within acceptable limits by meeting with the safety functions. Deterministic safety …"	Syntax error: Current text places at the same level safety functions and mitigating systems as two objects of the same nature. However, mitigating systems ultimate goal is indeed to prevent radiological releases by helping meeting with safety functions. For instance, removing of residual heat can be performed by feed and bleed by means of the HPIS and PORV(PZR), where the needed systems are the pumps and the relief valves.			X	The existing text seems clear enough and it seems not necessary to change it.
Canada 6	2.3	2.3 Deterministic safety analyses predict the response to postulated	It is not clear what is meant by "…postulated			X	These additional failures can be

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		initiating events ~~possibly combined with additional postulated failures~~ as well as consequential failures associated with the event and, for DBA, single component failures in accordance with the single failure criterion (7.35). A set of rules …	initiating events possibly combined with additional postulated failures." Is this referring to consequential failures, random failures, or something else entirely?				consequential failures or postulated single failures or postulated multiple failures (DEC). Paragraph 2.1 seems not to be the adequate one for these details (see Section 7 of the Safety Guide).
Russia 1 Comment 3	2.4	2.4. The results of computations are spatial and time dependent values of various physical variables (e.g. neutron flux; ~~thermal~~ heat power of the reactor; pressures, temperatures, flow rates and velocities of the primary coolant; loads to physical barriers; concentrations of combustible gases, physical and chemical compositions of radionuclides, status of core degradation or containment pressure, source term to the environment and others).			*Para 2.4 will be modified as follows:* 2.4 The results of computations are spatial and time dependent values of various physical variables (e.g. neutron flux; reactor thermal power ~~of the reactor~~; pressures, temperatures,…		The term "reactor thermal power" seems widely used.
Turkey 10	2.5	-	Various types of acceptance criteria (design criteria, operational criteria, safety criteria) are defined on this paragraph and on the document. But there are no numerical or physical or any other type of real value is not presented.			X	It is a common practice that each State defines values for these criteria by themselves. However, some orders of magnitude and some examples are provided in the Safety Guide.

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			The document needs to be enlarged by giving particular examples and real values for those types of criteria for providing guidance.				
UK 3	2.5	:…The acceptance criteria can be expressed either in general, qualitative terms or as quantitative limits. Three categories of criteria can be recognized: a) safety criteria: these are criteria either directly related to the consequences of operational states or accident conditions or to the integrity of barriers against releases of radioactive materials; (b) design criteria: design limits for individual systems, structures and components, which are part of the design basis as important preconditions for meeting safety criteria; and (c) operational criteria: these are rules to be followed by operator during normal operation and anticipated operational occurrences, they provide preconditions for meeting the design criteria and ultimately the safety criteria."	The current arrangement is confusing to the reader and not quite accurate. The revised text is intended to make the hierarchy clearer.	X	*Both formulations seem equivalent and acceptable. 2.5 will be modified:* (b~~a~~) ~~d~~Design criteria: design limits for individual structures, systems ~~structures~~ and components, that ~~which~~ are part of the …meeting safety criteria ~~in the two following categories~~ (see Requirement 28 from SSR-2/1 (Rev. 1) [1]; and (c~~b~~) ~~o~~Operational criteria: these are rules …and AOOs;~~,~~ they provide preconditions for meeting the ~~need to be consistent with~~ design criteria and ultimately the ~~provide preconditions for~~		

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					~~meeting~~ safety criteria.		
Russia 3 Comment 5	2.5.	Difference among acceptance criteria, targets factor of deterministic safety analysis and safety criteria shall be represented in this Safety Guide	2.5. **Acceptance criteria** are used in deterministic safety analysis for judgment of acceptability of the demonstration of safety of a nuclear power plant…. 2.6. In this Safety Guide only the acceptance criteria that are the targets of the deterministic safety analysis are addressed.			X	(No specific proposal is made). See Resolution to UK-3 about this para.
Turkey 24	2.6 Line 2	"…analysis are addressed. These acceptance criteria, as approved by ~~T~~the regulatory body, may…:	Better expression.		*Paragraph 2.6 will be modified as follows:* 2.6. In this Safety Guide only the safety acceptance criteria that are the targets of the deterministic safety analysis are addressed. These acceptance criteria, as approved by ~~T~~the regulatory body, ~~may decide to approve acceptance criteria that~~ may include margins with respect to safety criteria"		
Canada 7	2.7 First line	2.7. In this Safety Guide, uncertainty analysis ~~are~~ is addressed in §6.21-	Editorial. Analysis is singular.	X			

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		§6.29.					
Turkey 25	2.7 Line 5	~~(b) Use of scaled experimental data;~~ (b) Use of data from scaled experiments;	Better expression.	X			
Turkey 11	2.8	-	Conservative and best estimate approaches: General assumptions are given on the document for these assumptions. But examples for the mentioned presented types of codes or computer analysis is also suggested to be given.			X	It seems not necessary to provide this type of examples (e.g. names of computer codes). Also, depending on the version, some codes are conservative in an old version but best estimate in a new one.
ENISS-2 (Observer / unsolicited)	2.8 Table 1	*Option "**4. Realistic**"* *Modify the last column ("Type of initial and boundary conditions") cell to add:* "Best estimate (or partly most unfavorable conditions)	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	Option 4 would no longer be "Realistic" if the change suggested is incorporated.
Canada 8	2.9 Last line	"…a way that the acceptance criteria would be met for all of them."	Editorial. Add "of" before them "them".	X			
Russia 2 Comment 6	2.9 second sentence	*Conservatism of computer codes is compelled and is introduced by codes developers for compensation of*	It is necessary to change this sentence for a more accurate explanation of the		*Para 2.8 will be modified as follows:* 2.8 Table 1 lists different options		Chapter 5 provides more explanations.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		simplifications of the models caused by shortcomings of calculation tools and insufficient knowledge of physical phenomena. In the early days these shortcomings were compensated with conservatism of computer codes.	reasons of conservatism in computer codes for safety analyses. Conservatism of computer codes is always compelled and is introduced by the code developers to compensate simplifications of the models caused by shortcomings of calculation tools and insufficient knowledge of physical phenomena. Earlier these shortcomings were significant and caused a big conservatism of computer codes.		currently available for performing deterministic safety analyses~~,~~ with different levels of conservatism associated with the computer code (see section 5), availability of systems and initial and boundary conditions for the analysis.		
Canada 9	2.10	Add to end of paragraph: "…in legacy analysis. Another use for Option 1 is for scoping analysis."	Conservative analysis still has a place, and often provides a cost effective way to address safety concern.			X	Conservative analysis is used for scoping analysis. However, para. 1.9 states that "this Safety Guide is devoted to the deterministic safety analysis for design or licensing purposes".
Russia 1 Comment 1	2.10	"…*Option 1 remains also in legacy analysis*…."	What legacy is meant here?	n/a		n/a	Only a clarification about the meaning of "legacy" is requested. : It refers to traditional analysis performed in the past
UK 4	2.11 Line 6	*Insert:* … to justify ~~conservative~~ selection of	It is not sufficient to represent uncertain			X	The proposed change seems not necessary.

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		input data and to reveal any chaotic effects requiring detailed study. Option 2 is commonly used…"	parameters in a single deterministic calculation. That is poor weather forecasting.				The sentence indicates that "The complete analysis requires use of sensitivity studies…", so not only a single calculation is referred.
India 4	2.11	"…Option 2 is commonly used for safety analysis of anticipated operational occurrences and design basis accident."	The dose assessment of AOO and DBA should follow same approach (only frequency changes for AOO and DBA)			X	This SG offers two options for AOO analysis (either realistic or combined) depending on the objectives.
IAEA/WES-4 (Unsolicited)	2.11	"…Option 2 is commonly used for design basis accidents and for conservative analysis of anticipated operational occurrences."	Editorial.	X			
France 2	2.12 Line 3	"…However, in order to ensure the conservatism required in analysis of design basis accidents the uncertainties need to be identified, quantified and ~~statistically~~ combined…."	Uncertainties shall be combined but there is no need to mention how. Statistical combination of uncertainties is not always conservative.			X	"Statistical combination" is a term commonly used and it seems not understood as "conservative combination"
UK 5	2.12 Line 3	"… and partially the most unfavorable, initial and boundary conditions. ~~However, in order to ensure the conservatism required in analysis of design basis accidents the uncertainties need to be identified, quantified and statistically combined~~ For example, the analysis needs to reflect the limiting operating criteria which the analysis substantiates, and to take sufficient account of modelling uncertainty to provide an appropriate	The text is too vague.		*Para 2.12 will be modified as follows:* 2.12. Option 3 is so called BE plus uncertainty approach. This ~~which~~ allows the use of a BE computer code together with more realistic hypotheses. ~~,~~ ~~that means b~~Best estimate and partially ~~most~~ unfavorable,		

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		level of confidence in the conclusions. Availability of systems…"			initial and boundary conditions may be used taking into account the very low probability that all parameters would be at their most detrimental value at the same time. However, …"		
IAEA/ WES-5 (Unsolicited)	2.12 Last sentence	"…Option 3 … is at present accepted for some design basis accidents and for conservative analyses of anticipated operational occurrences"	Editorial.	X			
Jordan 1	2.14	It is suggested to add a figure (e.g. see Figure 1) to illustrate the decreasing level of conservatism from options 1 to 4.	In order to illustrate the decreasing level of conservatism from options 1 to 4			X	*(Note: see Fig 1 at the end of this Table).* Proposed illustration seems not necessary. Table 1 seems enough.
Russia 2 Comment 7	2.15 second sentence	2.15. Option 4 may be appropriate for realistic analysis of anticipated operational occurrences aimed at assessment of control system capability (§7.17-§7.44) and in general for best estimate analysis of design extension conditions (§7.45- §7.67) as well as for the realistic analysis with the purpose of justification of the operator's actions (§7.68 - -§7.6x).	To complement this sentence with words: "as well as for the realistic analysis with the purpose of justification of the operator's actions".		*Para 2.15 will be modified as follows:* 2.15. Option 4 may be appropriate …for BE analysis of DEC (7.45-7.67) as well as for the realistic analysis with the purpose of justification of the operator's actions. Deterministic…"		Referred paragraphs 7.68-7.6x are about practical elimination and not about justifying operator actions.
IAEA/WES-6 (Unsolicited)	2.18 Line 1 And	"2.18. The Source term is evaluated for operational states and accident conditions for the following reasons: …	Ensuring consistency with the terminology, concepts and approaches established in the Draft Safety Guide	X	*Item (g) will be modified as follows:* (g) To support safety the design of		The term "severe accident" is used in the Safety Glossary

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
	Item (g)	(g) To support ~~safety~~ the design of ~~mitigating~~ safety features and safety systems ~~related to~~ for the mitigatory domain of severe~~-~~ accident management (e.g. ~~F~~filtered ~~C~~containment ~~V~~venting~~)~~, hydrogen passive autocatalytic recombiners) [18].”	DS483 “Severe Accident Management Programmes for Nuclear Power Plants” (revision of NS-G-2.15), see e.g. Para 1.6 and Table 1 therein. A new reference [18] to DS483 should be inserted in bullet (g). Source term studies are also important for the design of hydrogen passive autocatalytic recombiners, another safety system for the mitigatory domain of severe accident management. A list of relevant references related to this issue can be provided by the reviewer upon request.		~~mitigating~~ safety features and safety systems ~~related to~~ for the mitigation of severe accidents ~~management~~ (e.g. ~~F~~filtered ~~C~~containment ~~V~~venting~~)~~ and recombiners of combustible gases~~, hydrogen passive autocatalytic recombiners~~; see NS-G-2.15 [11]).		
Turkey 12	2.18	The paragraph is suggested to be removed. Alternative: _____ Determination of the Emergency Action Levels (EALs) defined in GSR Part 7 and GSG 2 should be addressed	Source term estimation is an integral part of the whole safety assessment which is addressed by the given criteria. Not exactly complying to the presented list _____ Explanation regarding determination of the EALs which is the backbone of			X	Para 2.18 gives current view on reasons why source term is evaluated. In the footnote (3) is stated that application and establishing of emergency arrangements are beyond the scope of this Safety Guide, referring to GSR Part 7 and to

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		among the emergency arrangements.	the triggering mechanism for initiation of emergency response is only given in GSG 2 among the recent documents of the IAEA. Furthermore, the link between the EALs and deterministic safety analyses is not provided. However, the EALs should be determined in the licensing stage before commissioning of the NPPs. Thus the importance of deterministic safety analysis in terms of determining EALs should be emphasized in this document. The details can be given in another document which is related to emergency preparedness and response.				other Safety Guides. *(Note from TO: The text indicated under "Reason" in Turkey 12 is repeated in Turkey 9 (general comment). Also in Turkey 17, although this comment is not included in the Table because no para. or comment is provided there, only the "Reason")*
Turkey 13	2.18 Line 3	"(a)…as low as reasonably achievable during normal operation."	Application of the ALARA principle to uncontrollable situations like AOOs & accident conditions would be unrealistic.			X	The source term should be minimized in all plant states.
Turkey 14	2.18 Line 13	(f) To provide data for training activities regarding emergency arrangements;	Needs to end with a semi-colon (;)	X			
Russia 3 Comment 4	2.18. (d)	(d) to confirmation that the design guarantees sufficiency of the measures	Taking into account IAEA Guides GSR part 7, GS-G-			X	Existing text seems accurate enough. The

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		on emergency response that are required to protect human life, health, property and the environment in case of an emergency at the NPP which were installed in IAEA requirements and recommendations (GSR part 7 [9], GS-G-2.1 [10], GSG-2 [11])".	2.1 and, GSG-2.				safety standards mentioned are indicated in the footnote (3).
Turkey 15	2.19	The paragraph is suggested to be removed.	The paragraph does not provide any additional information.			X	As the para. seems convenient for other reviewers it is left untouched.
Jordan 2	*New 2.20 (or after section 8)*	Suggestion to insert section 2.20 or a Section after section 8 to identify or summarize the procedure used in this document describing how to perform deterministic Safety analysis. The proposed procedure as concluded from the document: 1. Identify and categorize initiating events. 2. Establish acceptance criteria. 3. Establish the analysis approach and select the proper computer code. 4. Develop the plant model. 5. Perform the analysis. 6. Compare the results with the relevant acceptance criteria. In addition, It is proposed to add a	To identify or summarize the procedure used in this document describing how to perform deterministic Safety analysis		*The last part of para. 1.16 will be modified as follows:* "…other sections of this Safety Guide; the sequence of these sections corresponds to the general approach, in terms of process, to perform deterministic safety analysis." *(Note: see Fig 2 at the end of this Table).*		

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		figure to illustrate this procedure (e.g. see Figure 2).					
Section 3							
Russia 1 Comment 8	Title Section 3 and Contents	3. IDENTIFICATION GROUPING AND CATEGORIZATION OF POSTULATED INITIATING EVENTS AND ACCIDENT SCENARIOS	This addition of a term from the current version of a document more correctly reflects the content and purpose of this Chapter		IDENTIFICATION, ~~AND~~ CATEGORIZATION AND GROUPING OF PIEs AND ACCIDENT SCENARIOS *(See para. 1.17)*		
Russia 1 Comment 9		Postulated initiating event (PIE) – it is a single failures of elements, or multiple failures of safety system elements, including common cause failures, which initiation in conditions of defense –in- depth specific level leads to transition into conditions of following level of defense –in- depth.	Taking into account targets of deterministic safety analysis which are enumerated in section 3 it is offered new term for postulated initiating event.			X	Postulated initiating event (PIE) is defined in the IAEA Safety Glossary (see "initiating event")
EC-JRC - 4 (Observer/ unsolicited)	3.2 Line 1	"…originated offsite or onsite ~~in any part of the plant~~ potentially leading..."	Clarification (external events belong to PIEs as well).			X	External hazards are not considered initiating events by themselves but give loads that may cause failure of reactor systems. It is the failure of reactor systems that is the "initiating event".
Canada 10	3.3 Line 2	3.3. Where applicable, it should be considered that a single cause can simultaneously initiate postulated initiating events in several or even all reactors, spent fuel storage~~s~~ and any	Editorial. "Spent fuel storages" should be singular.	X			

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		other …					
ENISS - 3 (Observer/ unsolicited)	3.3	3.3. Where applicable, it should be considered that ~~a single cause~~ specific hazards can simultaneously ~~initiate postulated initiating events in~~ give rise to impact on several or even all reactors, spent fuel storages and any other sources of potential radioactive releases on the given site (SSR 2/1 (Rev. 1), §5.15B) [1].	Para 5.15B of SSR 2/1 refers to "specific hazards give rise to impacts on several or even all units on the site simultaneously." (see also para 3.52)			X	*See EC/JRC-4.* External hazards are not considered initiating events by themselves but give loads that may lead to PIEs.
Canada 11	3.4 Line 3	"…Initial conditions should consider a stationary state with normal operation equipment operating prior to the initiating fault."	Editorial. Insert "a" before "stationary state".	X	*Second sentence will be modified:* "…Initial conditions should consider a controlled plant mode ~~stationary state~~ with normal operation equipment operating prior to the initiating event~~fault~~."		
Canada 12	3.4	3.4. The deterministic safety analysis should ~~be performed~~ address postulated initiating events that can occur in all planned modes of the plant during normal operation at full power and low power, including operation during shutdown. Initial conditions should consider stationary state with normal operation equipment operating prior to the initiating fault.	This change in wording allows for the possibility that explicit analysis of some operating modes can be avoided if it can be shown that analysis of another operating mode bounds the event for the mode in question.	X	*First sentence will be modified:* 3.4. The DSAshould ~~be performed for~~ address PIEs that can occur in all ~~planned~~ modes of ~~the plant during~~ normal operation ~~at full power and low power, including operation during shutdown~~		
UK 6	3.4	*Initial conditions should consider stationary state with normal operation*	The proposed text is not adequate as a requirement.			X	The change could have sense although its

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		equipment operating prior to the initiating fault. Add: "… initiating fault unless there is reason to believe that the fault occurrence is more likely during operational transients, or that the plant's mode of operation anticipates regular operational transients."					implementation seems unpractical. Actually it is not reasonable to perform multiple analyses for a given PIE considering any possible control systems configuration during normal transient. Stationary state of the plant is not real but a convention.
IAEA/WES-7 (Unsolicited)	3.5	3.5 Every configuration of shutdown modes including refueling and maintenance should be considered. For these modes of operation, contributors potentially increasing risk should be considered, such as: (a) the inability to start some safety systems automatically or manually; (b) disabled automation systems; (c) equipment in maintenance or in repair; (d) reduced amounts of coolant in the primary circuit as well as in the secondary circuit for some modes; (e) instrumentation switched off or non-functional and measurements not made; and (f) open primary circuit and open containment.	With a view to support structuring and to improve the readability of the entire sentence, please include consecutive numbering of the individual contributors potentially increasing risk in shutdown modes.			X	Numbering could seem to imply that the list is complete.
Turkey 26	3.6 Line 1	3.6. For postulated initiating events ~~initiated in~~ connected with the spent fuel pool, specific operating modes related to…"	Better expression.		*Para 3.6 will be modified as follows:* "… PIEs ~~initiated in~~ related to the spent fuel pool…"		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
India 5	3.6 Line 1	"…For postulated initiating events initiated in the spent fuel pool or during spent fuel transfer from core to pool, specific operating modes related to."	During fuel transfer also, some events may happen like failure during dry transfer or fuel handling/transfer machine cooling so highlighted text may be added		*See the Resolution to Turkey 26.*		
France 4	3.7 After the end	3.7. Postulated initiating events potentially taking place … an early radioactive release or a large radioactive release. Nevertheless, the need to prevent or mitigate these events with appropriate procedures or means should be addressed on a case by case basis.	These events should be nevertheless prevented or mitigated by specific procedure or means if they could lead to an early radioactive release or a large radioactive release.	X	*The sentence proposed will be added. Additionally, the first sentence will be modified as follows:* 3.7 PIEs ~~potentially~~ taking place during plant operating modes with negligible duration in time may ~~not be considered~~ be excluded from the DSA after careful analysis…" *(See Turkey 27 below).*		
Czech Republic 1	3.7	*We propose to delete this article.* ~~3.7. Postulated initiating events potentially taking place during plant operating modes with negligible 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.~~	This article in its current form enables to avoid analyzing some design basis accidents in shutdown modes. The low contribution to overall risk to large or early radioactive release is not sufficient reason to avoid this analyzes due to the severity of their		See India 6	X	An assessment of the overall risk includes consideration of the consequences as well as the likelihood.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			consequences.				
India 6	3.7	~~3.7 Postulated initiating events potentially taking place during plant operating modes with negligible duration in time may not to 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~~	Although contribution to overall risk may be very low but all events should be analyzed or it should be shown that PIEs enveloping the short term PIEs are already taken care of.		See Czech Rep 1	X	As it seems not possible to analyse all events it seems to make sense to exclude those of low risk.
EC-JRC - 5 (Observer/ unsolicited)	3.7 Line 4	"… large radioactive release (see §3.55)"	'Early radioactive release' and 'large radioactive release' are tricky terms that can raise ambiguity or controversy. This is why whenever they are used along the text, they should be put into commas and/ or accompanied by the appropriate reference.			X	No commas are used in SSR-2/1 (Rev.1)
Turkey 27	3.7 Line 2	"…duration in time may not be considered for deterministic safety analysis after careful analysis and quantitative assessment of its…:"	Better expression.		*See Resolution to France 4.*		
Pakistan 1	3.9 new bullet after h	(Add new point) (h1) Mid-loop operation	Shutdown PSA evaluation has established that mid-loop operation leads to high Core Damage Frequency (CDF) considering following scenarios;			X	Mid-loop operation is specific to certain reactor types; it seems better not to include it.

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			« Loss of AC power and loss of RHR during mid-loop operation • Loss of RHR pump due to mechanical failure during mid- loop operation				
EC-JRC - 6 (Observer/ unsolicited)	3.9 Line 3	"…typically include ~~conditions~~ plant operating states such as:"	Precision		*The sentence will be modified:* "…typically include operating conditions such as:"		
India 7	3.10 2nd line	"…parameters are changing due to the transfer to different plant modes or the changes in the plant power ~~output~~ or different core flux modes…"	Core flux mode may be flat or peak mode, initial, pre-equilibrium, equilibrium core conditions, which are important for analysis			X	"Different core flux modes" seem adequately covered by "different plant modes".
Canada 13	3.13 Line 1	3.13. The list of postulated initiating event~~s~~ should take due account of …	Editorial. "Event" should be plural.	X			
Turkey 28	3.13 Line 2	"…feedback~~,~~. This includes, depending on availability of relevant data, operating experience from the…"	Better expression.		*Line 2 from para. 3.13 will be modified as follows:* "…feedback, which ~~this~~ includes, depending on …"		
ENISS - 4 (Observer/ unsolicited)	3.14	"3.14. The set of postulated initiating events ~~should be comprehensive and~~ should be defined…"	As it is already recommended in para 3.12, it is not necessary to repeat in para 3.14 that the set of PIE should be comprehensive	X			
Turkey 29	3.14	"• failures initiated by operator errors,	Better expression.	X			

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	Line 5 Bullet 2	which ~~this~~ could range from faulty or incomplete maintenance…"					
Turkey 2	3.15 Bullet 3	*Please, explicitly state that this expression applies for any equipment of any system.*	Clarification to ensure that the statement does not leave any question mark in mind; such as "Could the failure of a protection division be counted inconsequential?"			X	Text seems sufficiently clear.
Turkey 16	3.15 Dash 2	Energetic event,	Example is given for the expression. However, the term is suggested to be explained more explicitly.			X	Text seems sufficiently clear.
Turkey 30	3.15 Line 11 Dash 2	"…potential failure of all equipment that can be affected by the consequences of the energetic event, such as the released hot water or the whipping pipe~~equipment which could be affected~~;"	Better expression.			X	The change proposed seems too detailed and technology dependent. Existing wording seems sufficiently clear.
Turkey 31	3.15 Line 14 Dash 3	**"…** equipment ~~which~~ that is neither designed to withstand the effects of the event nor protected from it."	Better expression.	X			
Canada 14	3.16	*First sentence, suggest to change:* "…(e.g. single failure criteria in DBA analysis)"	SFC does not apply to AOO and BDBA	X			
Canada 15	3.16 Line 1 and 2	3.16. Additional failures are assumed in deterministic safety analysis for conservatism (e.g.**,** single failure criteria) or for the purpose of defence in depth (e.g.**,** common cause failure). Distinction should be made between these additional failures and failures	Editorial. Add commas ",". The abbreviations *etc.*, *i.e.*, and *e.g.* are parenthetic expressions and should be enclosed between commas. Note that such editorial, if adopted, should apply to	X (for)		X	Use of comma after "e.g.": IAEA internal editorial rules will be followed.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		that are part of, or directly caused by, the postulated initiating event. Further failures may be added to bound a set of similar events, limiting the number of analyses. (e.g., single failure criteria) (e.g., common cause failure)	all occurrences of parenthetic expressions in the remainder of the text. Add "for" before "the purpose".				
IAEA/WES-8 (Unsolicited)	3.16 Line 2	"3.16. Additional failures are assumed in deterministic safety analysis for conservatism (e.g. single failure ~~criteria~~ criterion) or the purpose of defence in depth (e.g. common cause failure)."	Editorial.	X			
Canada 16	3.17 Lines 1 and 5	3.17. The postulated initiating events should only include those failures … hazards should be considered a potential cause of postulated initiating events, which includes resulting multiple failures.	Editorial. Line 1, "event" should be plural. Line 5, Change "include" to "includes" to agree with "a potential cause".	X			
Canada 17	3.17 Line 4	"…However, the loads associated with…"	Editorial. Add "," comma after "However". Some modern writers tend to drop the comma, but it is not a standard practice yet.	X			
Turkey 32	3.17 Line 4	"…not be considered as postulated initiating events by themselves. However the loads associated with…"	Grammar.	X			
UK 7	3.17 Line 3	Replace text beginning "therefore hazards" with "Internal and external hazards (natural and human induced) can potentially	Internal and external hazards should be treated as initiating events since they do challenge safety functions (indeed –			X	Text seems correct. It is the failure of plant equipment that can lead to releases of radioactive material. The loads and therefore

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		challenge multiple safety functions simultaneously and so need to be considered as postulated initiating events for which design provision such as barriers need to be provided to protect structures, systems and components against the loads associated with the hazard."	potentially more than one) for which the design needs to provide explicit design basis protection such as barriers to protect the plant against the loads associated with the hazard. The text seems to assume that these barriers already exist.				failures induced by the hazards are considered. Hazard by themselves are not considered as PIEs, see SSR2/1 (Rev. 1) para.5.6
Czech Republic 2	3.18	Combinations of independent events should be considered if it is not practically eliminated. This combination should be considered to be design basis accident or design extension condition depending on severity of its consequences and frequency of occurrence	The text is not clear and it can lead to misunderstanding.		*Para 3.18 will be modified as follows:* 3.18. Where the results of engineering judgement, deterministic ~~safety~~ assessments and probabilistic ~~safety~~ assessments indicate that …"	X	It seems better not categorize events into plant states based on consequences, as is suggested in the proposal.
Russia 2, Comment 8	3.18	3.18. Where the results of engineering judgement, deterministic safety assessments and probabilistic safety assessments indicate that combinations of independent events could lead to anticipated operational occurrences or to accident conditions, such combinations of events should be considered to be design basis accidents or should be included as part of design extension conditions, depending mainly on their complexity and	This para repeats the provision of the IAEA standard SSR-2/1 about a combination of events and failures stated in §5.32 which, in essence, cancels the principle of single failures as a basis for the analyses of design basis accidents, and replaces it with a combination of failures with the			X	Description of practices in individual States is out of the scope of this Safety Guide. The clarification indicated in the footnote proposed could apply to many other paras.

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		frequency of their occurrence. Footnote: Provisions of this para are adopted not in all member states.	established probability. This situation is unacceptable, as it can strongly increase the volume of calculations needed for search of the corresponding combinations. In Russia it is not accepted and for the analyses of design basis accidents only those combinations of events or failures are considered which are specified in the Federal norms and rules. In this regard it is necessary to make a footnote to this para and to specify there that provisions of this para are adopted not in all member states. Identification of the combination of failures getting to the range of the probabilities established for design basis accidents is a vulnerability which is subject to elimination in compliance with the				

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			practice of elimination of the vulnerabilities revealed by the probabilistic safety analysis.				
Canada 18	3.20 Line 1	3.20. Certain limiting faults (e.g., large break loss of coolant accidents,…	Editorial. Add commas “,”. The abbreviations *etc.*, *i.e.*, and *e.g.* are parenthetic expressions and should be enclosed between commas (see comment 59).				See Resolution to Canada 15. IAEA internal editorial rules will be followed.
Canada 19	3.21 Line 3	3.21. Failures occurring in the supporting systems that impede the operation of systems necessary for normal operation should be also considered as postulated initiating events if such failures eventually require the actuation of the reactor ~~protection~~ safety systems.	“Reactor protection system” is normally used just for equipment triggering reactor shutdown. The sense here is broader and includes all safety systems.	X	*The sentence will be modified as follows:* “… actuation of the reactor protection systems or safety systems.”		
Canada 20	3.22 Last line	3.22. The set of postulated initiating events should be reviewed as the design and safety assessments proceed and should involve an iterative process between these two activities. The postulated initiating events should ~~be~~ also be periodically reviewed throughout plant life to ensure that they remain valid, for example as part of a periodic safety review ~~to ensure that they remain valid~~.	Reword for clarity.	X			
Canada 21	3.23 Line 4	“…Therefore, they can be bound by a single…”	Editorial. Add “,” comma after “Therefore”. Some	X			

COMMENTS BY REVIEWER				RESOLUTION			
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			modern writers tend to drop the comma, but it is not a standard practice yet.				
Turkey 3	3.23 Last sentence	"…This approach allows the selection of the same acceptance criteria and initial conditions and the application of the same assumptions and methodologies to all postulated initiating events ~~grouped under the same representative event sequence~~ leading to the event sequences categorized under the same plant state**.**"	Specific acceptance criteria, initial conditions, and assumptions and methodologies are considered for each of the groups of postulated initiating events <u>depending on the frequency of initiating event</u>.		*(See also Turkey 33 about para. 3.24)* *Last sentence of 3.23 will be modified:* "… event sequence. As an example, the PIEs "stop of a Main Feed Water (MFW) pump", "stop of all MFW pumps" and "isolable break on MFW system" are all typically grouped under a single representative event sequence such as "Loss of MFW". *Paragraph 3.24 will be also modified:* "3.24. Representative event sequences … thermal shocks. ~~As an example, the postulated initiating events "stop of a Main Feed Water (MFW) pump", "stop of all MFW pumps", "isolable break on MFW system" are all typically grouped under a single~~	X	The change proposed seems not consistent with the paragraph.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					~~representative event sequence such as "Loss of MFW" which belongs to the "Decrease in reactor heat removal" type of event sequence~~ In the example above (para. 3.23), the representative sequence "Loss of MFW" ~~the representative sequence~~ belongs to the "Decrease in reactor heat removal" type of event sequence.		
EC/JRC-7 (Observer/ unsolicited)	3.24 All	Remove	Representative event sequences should stand for PIEs sharing similar threats to barriers or safety functions and necessary mitigating systems as suitably set forth in 3.23. 3.24 begins by stating that such representative sequences "can <u>also</u> be grouped" by type of sequences on reduced core cooling or containment pressurization. However, this does not constitute an additional criterion but an		*See Turkey 3*	X	A representative event sequence is already a group of PIE (as described in updated paragraph 3.23). But the representative sequences can be further grouped into types of event sequences that pose similar challenges to the barriers.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			example of a jeopardized critical safety function or barrier as a grouping criterion already mentioned in 3.23 (and 'decrease in heat removal' taken in the LFW example as well).				
Turkey 33	3.24 Line 5	"… all typically grouped under a single representative event sequence ~~which is~~ such as "Loss of MFW" …"	Better expression.		*See Turkey 3*		
UK 8	3.25	*Add:* • "Faults occurring while the reactor is a shutdown state."	Suggested because consideration of faults occurring while a reactor is in one of many of its shutdown states are often omitted from many safety analysis reports.			X	Initiating events are typically independent of the plant operating mode at the time of the event and the safety analysis reports should include events on a shutdown reactor state. However, this is not the correct section to address the proposal.
Russia 1 Comment 10		"… • Anomalies in reactivity and power distribution in the reactor core or in the fresh or spent fuel storage; • Anomalies in spent fuel management and violation of conditions of its cooling; • Anomalies in management of fresh fuel; • Increase or decrease of the reactor coolant inventory;…"				X	The first new bullet proposed seems sufficiently covered by existing bullets 3 and 7. The second seems adequately covered by existing bullet 3.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Russia 3 Comment 6	3.25.		Two similar sequences are brought in p. 3.25.: • Increase or decrease of the reactor coolant system flow rate; • Increase or decrease of flow coolant in primary circuit			X	The second bullet/sequence indicated in the comment (*Increase or decrease of flow coolant in primary circuit*) is not part of the list of bullets of this para.
Turkey 34	3.26 Line 4	"…could bypass the containment, because of potentially large consequences even in the case of relatively…"	Needs a comma.			X	The proposed comma seems not necessary
Finland 2	3.27 Line 2	"…depending on the bounding ~~total~~ frequency of the associated postulated initiating events…"	para 3.27 is not consistent with 3.36			X	Treated with and covered by "Canada 22"
Canada 22	3.27 Line 2	… depending on the ~~total frequency of the associated postulated initiating events~~ frequency of the most frequent postulated initiating event in the group.	• Use of "total frequency" is wrong for categorization of events into plant states. It could encourage subdivision of groups of events to lower the group frequency. This would allow inappropriate recategorization of some DBA as DEC, for example.	X			
France 3	3.27 Table 2	*In Table 2, for limiting faults, the lower bound of the indicative frequency range should not be mentioned:* « 1E-4 > f ~~> 1E-6~~ »	Frequencies of events in the range of 1E-6 are difficult to evaluate and an indicative frequency lower than 1E-6 is not always a sufficient reason to exclude an accident from		*3.20 will be modified to provide background reasoning:* 3.20. Certain limiting faults (e.g. large break loss of coolant accidents, main steam	X	Proposal to delete the lower limit: If no lower bound is given, then the frequency range would be unbounded and all events less frequent than 1E-4 would be included.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			DBC. Therefore it is better not to mention a lower bound for DBC-4. Moreover, it is more coherent with article 3.30 to remove this threshold.		or feedwater ~~secondary system~~ pipe breaks~~,~~ and control rod ejection in PWRs or rod drop in BWRs) are traditionally considered in DSA as DBAs. These accidents should be considered because they are representative of a kind of risk the reactor has to be protected from. They should not be excluded from …" *Additionally, a footnote will be added in 1E-4>f>1E-6(*):* "(*) Some other accidents which frequency is lower than 1E-6 should be considered because they are representative of a kind of risk the reactor has to be protected from"		
Czech Republic 3	3.27	Anticipated operational occurrences f>1E-1 Design basis accidents	The frequency ranges in TABLE 2 is inapplicable to existing nuclear power			X	*See France 3*

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		– infrequent faults 1E-1>f>1E-3 – limiting faults 1E-3>f>1E-4 – Design extension conditions 1E-4>f>1E-6	plants and difficult to achieve for present reactor designs. We propose to use different ranges or at least return to frequency range given in present SSG-2.				
Japan 1	3.27 TABLE 2 Footnote 4	[4] DBC: Design Basis Condition; PC: Plant Condition, in addition DBC-1 and PC-1 means normal operation.	Clarification. DBC-2, DBC-3, DBC-4, PC-2, PC-3 and PC-4 are only used here. Need to clarify the definitions in the footnote.	X	*Footnote will be modified:* DBC: Design Basis Condition; PC: Plant Condition; (DBC-1 and PC-1 are used for 'normal operation')		
Canada 23	3.27 and Table 2	**This is a major comment.** Delete last sentence of 3.27 and table 2. Alternatively, other member states should be invited to provide their own accident category tables.	Table 2 should be deleted. The SSG-2 Rev. 0 had a DBA lower limit of 10^{-4}. SSG-2 Rev 1 includes a category of DBA called limiting faults with a frequency down to 10^{-6} per year (see Table 2). This is potentially very problematic for existing plants. Standards for external hazards typically use a lower frequency limit of 10^{-3} or 10^{-4} per year. The 10^{-6} per year example is seriously inconsistent with this.		*Treated with "France 3"*	X	

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			The abbreviations (DBC-3, PC-2, etc.) are not understood outside the countries of origin.				
Jordan 3	3.28 Dash 2	Change the term "control malfunctions" to "pressure control malfunctions". - Decrease in reactor heat removal: feed water pump trips; reduction in the steam flow rate for various reasons (pressure control malfunctions, main steam valve closure, turbine trip, loss of external load, loss of power, loss of condenser vacuum);	In order to clarify the type of control malfunctions leads to reduction in the steam flow rate.			X	A number of different control malfunctions could lead to a reduction in steam flow, e.g. incorrect load set point. (The proposed change only applies to specific reactor designs).
Jordan 4	3.28 Dash 4 (Page 15 Line 1)	Add another example of PIE to the list of PIEs lead to "decrease in reactor coolant flow rate" such as "Emergency deviation of the grid frequency".	For more clarification		*Regarding grid frequency, dash 3 will be modified as follows:* - Decrease in reactor heat removal: feed water pump trips; … turbine trip, loss of external load and other external grid disturbances, loss of power, loss …"		
Spain 1	3.28 and 3.30		Traditionally the "inadvertent operation of emergency core cooling" event has been considered as an anticipated operational occurrence.	X	*In 3.30, dash 5,* "inadvertent operation of emergency core cooling" *will be removed and placed*		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			Should this event be included in paragraph 3.28 instead of in paragraph 3.30?		*as AOO, under "*Increase of reactor coolant inventory*" (para. 3.28, dash 8).* *See also Jordan 5 comment on 3.30.*		
Turkey 35	3.28 Line 23 Dash 10	— Reduction or loss of fuel cooling in the ~~:~~ fuel pools: loss of off-site power; malfunctions in…	Unnecessary colon (:) needs to be deleted.	X	*(Also in Canada 24)*		
IAEA/WES-9 (Unsolicited)	3.28 Line 23	*Penultimate bullet:* "- Reduction or loss of fuel cooling in the~~:~~ fuel pools: …"	Delete the redundant colon.	X	*(Also in Canada 24)*		
India 8	3.28 Bullets 7 and 11	**Clarification** required as following appear in both AOO and DBA list : Loss of moderator circulation or decrease or loss of moderator heat sink (in pressurized heavy water reactor) Release of radioactive material due to leak in reactor coolant system, with potential containment bypass, or from a subsystem or component: minor leakage from a radioactive waste system.	These are repeated in the list of AOOs as well as DBA.	n/a	n/a	n/a	Requested clarification: They could be AOO or DBA depending on reactor design or other considerations, e.g. partial loss is AOO and complete loss is DBA.
Armenia 1	3.28 Line 13	"…inadvertent control rod/control rods bank withdrawal"	In most reactors control rods are moved within one bank and special measures should be taken to move a single control rod. So, the movement of a control rod		*Dash 5 from para. 3.28 will be modified as follows:* "…inadvertent control rod (or control rod bank) withdrawal"		

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			bank is more probable and the consequences are more severe.				
Canada 24	3.28 Item 10	— Reduction or loss of fuel cooling in the~~:~~ fuel pools: loss of off-site power; malfunctions in decay heat removal system; leaking of pool coolant; ~~malfunctions of the ventilation system;~~	Malfunctions of ventilation systems do not belong in this group of examples because they are not a loss of cooling. Editorial. Remove colon after "the" in first line.	X			
UK 9	3.28	Insert: Events with a frequency such that one such event is reasonably expected to occur within the life of the plant are termed anticipated operational occurrences. The expectation is that the plant will be designed such that the statutory environmental release limits are not expected to be breached during operation. In practice this is usually met by design criteria which require demonstration of the continued integrity of all barriers to release.	This is not adequately explained.			X	AOO is defined in the IAEA Safety Glossary. Further detail is provided in Chapter 7.
Canada 25	3.29 Line 2	"…All postulated initiating events identified as initiators of anticipated operational occurrences ~~should~~ may also be analyzed…"	If AOO is performed using DBA rules and it can be demonstrated that the AOO acceptance criteria can be met, no need to do AOO Level 2 DiD analysis.			X	AOO should [always] be analyzed with DBA rules to demonstrate the effectiveness of the protection system. It is realistic analysis of AOO to demonstrate effectiveness of the control systems that

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							may not be necessary.
Turkey 36	3.29 Line 5	"…of occurrence, the ~~establishment~~ determination of any threshold limit should consider the safety …"	Better expression.			X	Threshold limits (such as the frequency band for AOO) are normally defined by regulators and usually based on [good] international practices. They are not determined as part of deterministic safety analysis.
UK 10	3.29	Insert: "Design basis accidents are events which, while not expected to occur, would result in an unacceptable level of risk if no protection was provided. The analysis of these faults seeks to justify the design of the protection systems and the operation criteria needed to ensure that the protection is functionally capable."	This is not adequately explained.			X	DBA is defined in the IAEA Safety Glossary. Further detail is provided in Chapter 7.
Armenia 2	3.30 Line 8	"…uncontrolled control rod/control rods bank withdrawal:"	See above		*See Armenia 1. Dash 4 from para. 3.30 will be modified as follows:* "…uncontrolled control rod (or control rod bank) withdrawal; …"		
Canada 26	3.30 Bullet 3	— Decrease in reactor coolant system flow rate: seizure or shaft break of main coolant pump; trip of all coolant pumps ~~(boiling water reactor)~~;	Trip of all coolant pumps is not specific only to BWR.	X			
India 9	3.30	New bullets may be added	These are important	X	*A new dash will be*		All the proposed

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		• End-shield Cooling failure • CV cooling failure • Main Steam Line break inside containment • Malfunction of support/auxiliary systems	type/phenomenon of PIEs. Malfunction of support / auxiliary systems may lead to AOOs/DBA		*added at the end of para. 3.30:* • End-shield cooling failure: (PHWR)		examples seem not necessary
Jordan 5	3.30 Page 15 Line 11	Provide Justification that inadvertent operation of ECCs can be considered as design basis accident. As in some practices in some countries is considered within Anticipated Operational Occurrences list.	In some countries, Inadvertent operation of ECCs is considered within Anticipated Operational Occurrences list.	X	*(See also Spain 1) Dash 5 from para 3.30 will be deleted:* - ~~Increase in reactor coolant inventory: inadvertent operation of ECC~~;		
ENISS - 5 (Observer/ unsolicited)	3.31	"3.31. Probabilistic analysis should be used as a support to justify the categorization of postulated initiating event according to their frequency of occurrence. In that case frequency calculation should consider the estimated duration of the relevant plant states to justify that an event occurring in shutdown states are not in the same category as the same event occurring during normal plant operation at power. It should specially be checked …"	Complementary information for the use of probabilistic analysis to justify the PIE categorization.		*A new second sentence will be incorporated:* "… of occurrence. The calculation of the frequency should take account of the relative frequencies of plant operational states according to its occurrence, such as full power or hot shutdown. It should specially be …"		
Canada 27	3.31 Line 2	3.31. Probabilistic analysis should be used as a support to justify the categorization of postulated initiating events according to their frequency of occurrence.	Editorial. Events is plural.	X			

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ENISS - 6 (Observer/ unsolicited)	3.35	"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, unless practically eliminated as presented in paragraphs 7.68 to 7.72 of this Safety Guide. They include: (a) A reduction in or loss of cooling of the fuel in the spent fuel pool (if leading to boiling) when the pool is located outside the containment; (b) Reactivity anomalies …"	In any case, all the listed situations should be considered unless they are practically eliminated. For example, 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	First change proposed: As para. 3.35 deals with AOOs and DBAs it seems not necessary to mention 'practical elimination'. The next three subsections deal with DEC and consider PIEs that are not practically eliminated. Second change proposed: AOOs and DBAs in the spent fuel pool would not be expected to lead to boiling.
IAEA/WES-10 (Unsolicited)	3.36	3.36. The frequency associated with a bounding event sequence belonging to an anticipated operational occurrence~~s~~ or a design basis accident should use the bounding frequency established for the postulated initiating events that have been grouped together."	Editorial.	X			
Turkey 37	3.37 all	3.37. In accordance with SSR-2/1 (Rev. 1), Req. 20 [1], design extension conditions ~~which~~ that are either more severe than a design basis accident or that involve additional failures should be identified using engineering judgement as~~,~~ well as deterministic and probabilistic assessment, with the objective of identifying design provisions to prevent as far as possible	Better expression.		*Para 3.37 will be modified as follows:* "… design extension conditions ~~which are either~~ more severe than a design basis accident or ~~that involve~~ involving additional failures, should be identified using engineering		

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		such conditions or mitigate their consequences.			judgement, as well as deterministic and probabilistic assessment, with the objective …"		
Russia 1 Comment 11	Title before para 3.37	GENERAL CONSIDERATIONS FOR IDENTIFICATION OF DESIGN EXTENSION ~~CONDITIONS~~ ACCIDENTS 3.31.	"design extension accident" is a regular definition. "design extension conditions" is a consequence of this definition.			X	"Design extension conditions" is the term used in SSR-2/1 (Rev.1) and in the Safety Glossary.
Russia 1 Comment 12	3.38	3.38. Two separate categories of design extension conditions should be identified: design extension conditions without ~~significant fuel degradation~~ violation of design limits of fuel damage and design extension conditions	"significant fuel degradation" is not an engineering definition. Design limits of damage are arranged for each type of fuel in each country.			X	Terms used in SSR-2/1 (Rev.1) and in the Safety Glossary.
UK 11	3.40 Bullet 1, line 2	Insert: "…multiple tube rupture where the manufacture has evidence to discount common-mode and consequential tube failures in a steam generator of a pressurized water reactor."	In general, this is not a good example, or even appropriate without a caveat.			X	The introductory text clarifies that these are lower frequency and more severe events than the equivalent DBA.
ENISS - 7 (Observer/ unsolicited)	3.40 Bullet 2 Line 5	"… The failures of supporting systems should be identified ~~are implicitly included~~ among the causes of failure of safety systems.	Consistently with para 3.42, it is recommended to identify explicitly DEC related to failure of supporting systems as it could lead the specific design requirement for diversified supporting functions			X	The meaning of the existing wording is that a failure of a supporting system that leads to the failure of a safety system is considered as one of the possible causes of failure of the safety system.

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IAEA/WES-11 (Unsolicited)	3.40 Bullet 2 Line 7	"… a total failure of any safety system credited in the safety analysis, for each anticipated operational occurrence~~s~~ or design basis accident (at least for the most frequent ones);"	Grammar.	X			
Turkey 38	3.40 End of bullet 2	"…frequent and/or challenging ones);"	Not only the most frequent AOOs and DBAs but also the most challenging ones need to be systematically analyzed.			X	Additional failures added to low frequency events can be considered as part of the residual risk due to very low frequency.
Turkey 39	3.41 Line 3	"…significant fuel degradation and to be adapted ~~plant specifically~~ to plant type and design:"	Better expression.		*Line 3 of para. 3.41 will be modified:* "…and ~~to be~~ specifically adapted to the type and design of the plant ~~specifically~~:"		
Canada 28	3.41 Bullet 2	"…anticipated operational occurrences or design basis accident combined with multiple failures ~~on~~ in safety systems"	Editorial. Change "on" to "in".	X			
Turkey 40	3.41 Lines 4 and 10	accident → accident~~s~~	Need to be plural.	X			
UK 12	3.41	Insert: Note steam demand faults with consequential tube rupture is likely to be a design-basis fault.	The example is misleading, because this sequence traditionally bounds much more frequent faults.			X	The introductory text makes it clear that these are lower frequency and more severe events than the equivalent DBA.
UK 13	3.41 Bullet 1, dash 1	*Delete uncontrolled boron dilution.* ~~• *uncontrolled boron dilution (PWR);*~~	This is frequent and therefore a design basis fault.		• uncontrolled heterogeneous boron dilution (PWR);	X	The introductory text makes it clear that these are lower frequency and more severe events than the equivalent DBA.

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					(Note: coolant intake due to several reasons, e.g. LOCA)		
UK 14	3.41 Bullet 3, dash 2	*Delete loss of residual heat removal.* • ~~total loss of core cooling in the residual heat removal mode~~;	This is frequent and therefore a design basis fault.		*The sub-bullet will be modified as follows:* - Loss of the ~~total loss of core cooling in the~~ residual heat removal system during cold shutdown or refuelling ~~mode~~;		
UK 15	3.41 Bullet 3, dash 4	*Delete loss of ultimate heat sink* ~~• loss of normal access to the ultimate heat sink~~	This is frequent and therefore a design basis fault.			X	The introductory text makes it clear that these are lower frequency and more severe events than the equivalent DBA.
Jordan 6	3.41 Page 18 Line 32	Add a note clarifies that Station Blackout accident can be considered as a design extension condition with core melting.	Station Black out accident can be considered as a design extension condition with core melting for old generations of NPPs (such as Gen 1 & 2). However, For the new generations (such as Gen 3 & 3+) is considered within the list of design extension condition without fuel degradation as these generations incorporate innovative passive safety features in their design.			X	The introductory text makes it clear that the examples do not apply to all NPPs. Also, it can be noted that a station blackout is not necessarily a DEC. Loss of all AC power with simultaneous loss of alternate AC power may be DEC – it depends on the duration of the postulated loss of AC power, which depends on the design.

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ENISS - 8 (Observer/ unsolicited)	3.45 Line 3	"…These sequences should be selected in order to represent all main physical phenomena (e.g. primary circuit pressure, reactor decay heat, containment status) involved in core melt sequences."	It is worth giving examples of the main physical phenomena involved in core melt sequence	X			
Turkey 41	3.45 Line 2	"…to establish the design basis for the safety features for mitigating the consequences of core melting accidents, according to…"	Better expression.	X			
IAEA/WES-12 (Unsolicited)	3.46 Line 4	"…responses to the design basis accident or design extension condition~~s~~ sequences and …"	Editorial.	X			
Turkey 42	3.46 Line 4	"…design basis accident or design extension condition~~s~~ sequences and ~~to~~ by considering the dominant accident…"	Better expression.		*Para 3.46 will be modified as follows:* "… core melting fail or are insufficient, and that an accident sequence … representative sequences should be selected by considering ~~adding~~ additional failures…"		
IAEA/WES-13 (Unsolicited)	3.47 Line 1	"Representative ~~sequences with core melt (~~design extension condition~~s~~ sequences with core melting~~)~~, regarding each ~~criteria~~ criterion, should be analysed to determine limiting conditions."	Wording; Grammar.	X			
Canada 29	3.47 Whole para.	3.47. Representative sequences with core melt (design extension conditions with core melting), regarding each	Paragraph is not clear and contains grammatical errors.	X	*Additionally to the changes indicated in IAEA/WES 13,*		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		criteria, should be analysed to determine limiting ~~conditions. Particularly, those~~ conditions, particularly those that could challenge containment integrity~~. Which~~ The representative sequences should be used to provide input to the design of the containment and of those safety features necessary to mitigate the consequences of such design extension conditions.			*para. 3.47 will be modified as follows:* "… determine limiting conditions, ~~.~~ ~~P~~particularly~~,~~ those that could challenge containment integrity. ~~which~~ The representative sequences should be used to provide input to …"		
Turkey 43	3.47 all	3.47. Representative sequences with core melt (design extension conditions with core melting) ~~regarding each criteria~~ should be analysed, regarding each criteria, to determine limiting conditions. Particularly, ~~those~~ sequences that could challenge containment integrity or those safety features necessary to mitigate the consequences of such design extension conditions should be used to provide input to the~~ir~~ design of the containment …"	Better expression.		*See resolutions to IAEA/WES 13 and Canada 29.*		
ENISS - 9 (Observer/ unsolicited)	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 normal access to the ultimate heat sink~~, a total loss of feed water with failure of core melt prevention feature;	The first example of DEC with core melt is not relevant for plants integrating lessons learned from the Fukushima accident. We suggest to replace it by a more relevant example			X	The suggested new formulation seems not better than the existing one.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
France 5	3.49	"…Core melt conditions should be postulated regardless of the provisions implemented in the design and the possibility of some very energetic phenomena that may result from the core melt accident should be prevented (i.e. the possibility of the conditions arising may be considered to have been 'practically eliminated') to exclude containment failure."	France supports this sentence that should not be modified.		*See Canada 30*		
Canada 30	3.49 Whole	3.49. The low frequency of occurrence of an accident with core melting is not sufficient reason for failing to protect the containment against the conditions generated by such an accident. Core melt conditions should be postulated regardless of the provisions implemented in the design. To exclude containment failure, ~~and the possibility of~~ the analysis should demonstrate that ~~some~~ very energetic phenomena that may result from ~~the~~ core melt accidents should be prevented (i.e. the possibility of the conditions arising may be considered to have been 'practically eliminated') ~~to exclude containment failure~~.	The paragraph gives guidance for design. This guide is about safety analysis. Reword to give guidance that is within the document scope.	X			
Russia 3 Comment 7	3.49	It is offered not to consider exposure of "practically eliminated" conditions on containment .	3.49. Core melt conditions should be postulated regardless of the provisions implemented in the			X	According to the footnote (2) in para. 2.11 from SSR-2/1 (Rev 1), that is the purpose of 'practical elimination': *"(2) The possibility of*

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			design and the possibility of some very energetic phenomena that may result from the core melt accident should be prevented (i.e. the possibility of the conditions arising may be considered to have been 'practically eliminated') to exclude containment failure. It seems that this requirement is impracticable as for antecedent generation NPP as well as for new power generation Units of NPP. Exclusion of containment failure is not possible within scenario which includes reaction of corium with architectural concrete, in condition with fuel melting under high pressure in reactor, as well as within steam explode or hydrogen explode.				*certain conditions arising may be considered to have been 'practically eliminated' if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be extremely unlikely to arise"*
Canada 31	3.50 Whole para	3.50. ~~Severe accident~~ Design Extension Conditions sequences should be selected to identify the most severe plant parameters resulting from the severe accident phenomena. These	Not all severe accidents are considered in design, so first occurrence should be changed to DEC. The paragraph gives		*Paragraph 3.50 will be modified as follows:* 3.50. ~~Severe accident~~ Representative		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		parameters should be considered in the ~~design of the plant structures, systems, and components~~ ~~that are necessary to limit~~ deterministic safety analysis that demonstrates the limitation of the radiological consequences of such severe accident se quences. The analysis should provide the environmental conditions ~~should~~ to be taken into account in the ~~qualification~~ assessment of equipment used in severe accidents.	guidance for design and equipment qualification. This guide is about safety analysis. Reword to give guidance that is within the document scope. Use of "qualification" in the final sentence implies full EQ. This is not required. Change to "assessment".		sequences of DECs with core melting should be selected to …These parameters should be considered in the deterministic analyses ~~design~~ of the plant SSCs ~~that are~~ necessary to demonstrate the limitation of the radiological consequences of such severe accident sequences. The analysis of these sequences should provide ~~T~~the environmental conditions ~~should~~ to be taken into account in the ~~qualification~~ assessment[(1)] on whether the ~~of~~ equipment used in severe accidents are capable of performing their intended functions when necessary (see Req 30 from SSR-2/1 (Rev.1) [1]). *Footnote (1): Although equipment qualification is out of the scope of this Safety Guide, it is understood that typical*		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					equipment qualification programmes for these accident conditions may not always be applicable and an assessment on the operability of SSCs is acceptable; according to that, the term "survivability assessment" is used in some States.		
Russia 1 Comment 13	3.50.	3.50. If a severe accident is characterized by low frequency of its occurrence but be hard consequences, this accident should be examined. Severe accident sequences should be selected to identify the most…				X	*See Resolution to Canada 31.* It seems better not to refer to frequency because in severe accidents the highest frequency is to be taken into account.
France 6	3.51	3.51. Determination of postulated initiating events should consider effects and loads from events caused by relevant site specific internal and external hazards (SSR-2/1 (Rev.1), Req. 17, §5.15A-§5.21A) [1]. A list of examples external hazards can be found in NS-R-3 [12]. 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 safety	The load case analysis could be complemented by an event case analysis. The previous sentence is sufficient to highlight that the hazards analysis differs from other ones.			X	The change would make the last sentence unnecessary. It seems better to keep the existing text for consistency with other paras and for greater clarity.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		functions6. The hazards ~~themselves do not represent initiating events but they~~ are associated with loads which can initiate such events.					
IAEA/WES-14 (Unsolicited)	3.51	"…A list of ~~examples~~ external hazards can be found in NS-R-3 (Rev. 1) [12]…."	In the frame of the IAEA Action Plan on Nuclear Safety, NS-R-3 was revised by amendment (DS462). NS-R-3 (Rev. 1) was published in February 2016. Please note that the current wording of the 2nd sentence is potentially misleading because NS-R-3 (Rev. 1) does not provide a separate table or list of external hazards; instead, Section 3 of NS-R-3 (Rev. 1) establishes specific requirements for the evaluation of external natural and human induced events.	X			
Canada 32	3.51 Line 3	A list of examples of external hazards can be found in NS-R-3	Editorial. Add missing "of".			X	*Covered by and treated in combination with IAEA/WES 14*
UK 16	3.51 Line 7 (last sentence)	Delete sentence being "~~The hazards themselves do not represent initiating events but they are associated with loads which can initiate such events~~."	See comment 7 above.			X	First part (to delete the sentence): see resolution to France 6 and IAEA/WES 14

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		And replace with: "Internal and external hazards represent initiation events for which design provision such as barriers need to be provided to protect structures, systems and components against the loads associated with the hazard."					Second part (to add the new text): the proposed para corresponds to design and is out of the scope of this Safety Guide. *See Resolution to France 6 and IAEA/WES-14*
IAEA/WES-15 (Unsolicited)	3.52 Line 1	"In accordance with SSR-2/1 (Rev.1), §5.15B, §5.19, and §5.63 [1] in determination of postulated initiating events caused by site specific hazards for multiple unit plant sites …"	The reference number is missing and should be inserted for completeness.	X			
France 7	3.53 and 3.54	3.53 The analysis of hazards[7] ~~which is performed by using probabilistic methods or appropriate engineering methods~~ should demonstrate that either: • such hazard can be screened out due to its negligible contribution to risk, or ~~• the nuclear power plant design is robust enough to prevent any transition from the load into an initiating event, or~~ • the hazard causes an initiating event considered in the design, ~~or~~ and the analysis ~~should~~ crediting only ~~credit~~ SSCs that are qualified or protected for the hazard is satisfactory ~~.~~ In any case, the protection of plant equipment provides guaranties that the hazard will not result in an accident condition	Consider deletion or complementary explanation why this article allow a choice between "probabilistic" and "engineering" Consider deletion: the difference between the first and second is not clear and the second sounds overconfident. "The hazard causes an initiating event" is not a recommendation.		*The last bullet of para 3.53 will be deleted:* ~~• the protection of plant equipment provides guaranties that the hazard will not result in an accident condition~~	X	*Existing text is consistent with SSR-2/1 (Rev.1).* In the change proposed there is an "excessive requirement" that hazards never lead to accident conditions

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		3.54. In cases where an initiating event is caused by a hazard, the analysis should only credit SSCs that are qualified or protected for the hazard.					
IAEA/WES-16 (Unsolicited)	3.53 Last bullet:	"…the protection of plant equipment provides guarant~~ie~~es that the hazard will not result in an accident condition."	Editorial.		*See the change indicated in the Resolution to France 7*		
Pakistan 2	3.53	3.53. The analysis of hazards which is performed by using probabilistic methods or appropriate engineering methods should demonstrate that either: • such hazard can be screened out due to its negligible contribution to risk, or • the nuclear power plant design is robust enough to prevent any transition from the single load or multiple coincident loads into an initiating event, or • the single hazard or multiple coincident hazards cause~~s~~ an initiating event considered in the design, or • the protection of plant equipment provides guaranties that the single hazard or multiple coincident hazards will not result in an accident condition	An important lesson learnt from Fukushima Daiichi accident is the absence of accident analysis results for multiple coincident hazards/ occurrences (earthquake followed by Tsunami leading to flooding of EDGs and their failure). Thus it has been realized that NPP industry may consider coincidental occurrences (combination of incidents) of multiple natural hazards leading to failure of SSCs important to safety (earthquake + tsunami + loss of all AC power).		*See the change indicated in the Resolution to France 7*	X	The concern indicated in the proposed change is addressed in para. 3.17

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Canada 33	3.53 Bullet 4	• "the protection of plant equipment ~~provides guaranties~~ guarantees that the hazard will not result in an accident condition"	Editorial. Simplify and use correct spelling of "guarantees".		*See the change indicated in the Resolution to France 7*		
Russia 1 Comment 4	3.54	3.54. In cases where an initiating event is caused by a hazard, the analysis should only credit SSCs that are qualified or protected for the hazard	There is no disclosing of abbreviation SSC in the document.	X	*All abbreviations will be expanded before publication.*		
France 8	3.56	1) Events that could lead to prompt reactor core damage and consequent early containment failure, for example: […] 2) Severe accident sequences that could lead to early containment failure, for example: […]:	Add "for example" to be more neutral to the technology.		*Text in items 1) and 2) will be modified:* "…containment failure, such as: […]		
France 9	3.56 Bullet 3	~~3) Severe accident sequences that could lead to late containment failure:~~ ~~a. Basemat penetration or containment bypass during molten core concrete interaction (MCCI)~~ ~~b. Long term of loss of containment heat removal~~	For these severe accidents, it should be possible to mitigate their consequences. Thus, category 3 should not be part of the list.			X	All the risks are to be practically eliminated. *See SSR-2/1 (Rev.1) para. 5.27*
Turkey 44	3.56, Line 14, 3.b	b. Long term ~~of~~ loss of containment heat removal	Better expression.	X			
Jordan 7	3.56 2) (b) Page 22	It is suggested to split the phrase "Large Steam Explosion" into "in-vessel steam explosion" and "Ex-vessel steam Explosion"	In order to specifically demonstrate that both events sequences, which could lead to an early radioactive release or a large radioactive release, are practically			X	The change proposed seems not necessary. Both in-vessel and ex-vessel steam explosions are included by the current wording.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			eliminated.				
Jordan 8	3.56 Page 22	It is suggested to add "in-vessel and ex-vessel re-criticality after the core melt" to the list of the event sequences requiring specific demonstration of their practical elimination.	In order to demonstrate that this event sequence, which could lead to an early radioactive release or a large radioactive release, is practically eliminated.	X	*A new item 6) will be added:* 6) In-vessel and ex-vessel re-criticality after core melting.		
France 10	3.57	~~3.57. Consequences of event sequences that may be considered to have been 'practically eliminated' do not need themselves to be deterministically analysed. Nevertheless, severe accident management guidance for "not postulated scenario" should be provided.~~	Propose to DELETE 3.57 because controversial. Notably with the sentence "Nevertheless, severe accident management guidance for "not postulated scenario" should be provided" is very ambiguous : does it means that SAMG is recommended for practically eliminated conditions ? How shall "not postulated scenario" be interpreted here ? Moreover, if 3.57 is not deleted, it should be complemented with guidance on the consideration of these situation with probabilistic approach in level 2 PSA and the analysis that is			X	*See Resolution to "Russia 2 comment 9"*

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			needed to identify the situation as a situation to practically eliminate considering its consequences.				
EC-JRC - 8 (Observer/ unsolicited)	3.57 Line 3	N/A	The "not postulated scenario" is still not clarified. One can easily assume that this one refers to the practically eliminated scenarios which do not have to be deterministically analyzed but when placing the term into quotation marks is like if it is referring to a predefined set of scenarios different than the 'practically eliminated' ones. An example may help clarify this issue, together with an example of its inclusion in SAMGs.	n/a	*See Resolution to "Russia 2 comment 9"*	n/a	
Canada 34	3.57 Last line	"…Nevertheless, severe accident management guidance for "not postulated scenarios" should be provided.	Editorial. "Scenario" should be plural.	n/a	*See Resolution to "Russia 2 comment 9"*	n/a	
Russia 2 Comment 9	3.57	…that may be considered to have been 'practically eliminated' do not need themselves to be deterministically analysed for development of severe	This para contains a contradiction. It is specified there that for scenarios which are		*Paragraph 3.57 will be modified as follows:* 3.57. Consequences		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		accident management guidance.	considered "practically eliminated" it is not required to make deterministic analyses. At the same time it is stated that an accidents management guidance has to be developed for them. We consider it impossible to develop an accident management guidance without their analysis. Therefore it is necessary either to acknowledge that the concept of "practically eliminated" for a set of scenarios of severe accidents is equivalent to the concept of hypothetical accidents existing before Chernobyl accident and not to provide any measures for them, or to indicate in para 3.57 the need of implementation of deterministic analyses of such scenarios for development of accidents management guidance.		of event sequences that may be considered to have been 'practically eliminated' ~~do not need themselves to be deterministically analysed~~ are not part of the DSA. However, DSA contributes to the demonstration that design and operation provisions are effective in the 'practical elimination' of these sequences (see paras 7.68 to 7.72). ~~Nevertheless, severe accident management guidance for "not postulated scenario" should be provided.~~ *(Last sentence is out of the scope of this Safety Guide)*		
Comments provided without indicating the specific applicable paragraph/s							

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Russia 3 comment 9		It is offered make an addition item: Deterministic comment 10safety analysis is intended for the efficiency justification of safety system which are assigned for execution of main safety functions in all conditions of power generating Unit of NPP provided by item.3.1, where each condition corresponds to only one of five levels of defense –in- depth.	In accordance with SSR-2/1 (Rev.1) by main means for safety ensuring is an using of principle of defense- in- depth (DDP)			X	Unclear comment and proposal
Russia 3 comment 10		Postulated initiating event (PIE) – it is a single failures of elements, or multiple failures of safety system elements, including common cause failures, which initiation in conditions of defense –in- depth specific level leads to transition into conditions of following level of defense –in- depth.	Taking into account targets of deterministic safety analysis which are enumerated in section 3 it is offered new term for postulated initiating event.			X	Unclear comment and proposal
Russia 3 comment 11		The full list of PIE shall be formulated for the following categories of PIE: Category 1 includes PIE with expected at operational event (PIE AOO – Anticipated operational occurrences), initiation of which in conditions level 1 of defense-in- depth leads to transition in conditions level 2 of defense-in- depth , where the special safety systems of normal operation execute the main safety functions; Category 2 includes PIE of design basis accident (PIE DBA), initiation of which in conditions of previous levels				X	Guidance regarding the identification, categorization and grouping of PIEs and accident scenarios is provided in Section 3 for the different plant states. The specific lists of PIEs are reactor type and design specific and cannot be compiled in this Safety Guide. On the other hand, each State decides about using the categorization of PIEs provided in this

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		of defense-in- depth leads to transition in conditions level 3 of defense-in-depth , where the basic safety systems of normal operation execute the main safety functions; Category 3 includes PIE of beyond design basic accident (PIE BDBA), initiation of which in conditions of previous levels of defense-in- depth leads to transition in conditions level 4 of defense-in-f depth , without of exceeding of the design limits for nuclear fuel damage in core and spent nuclear fuel in fuel pools , where auxiliary safety systems of normal operation as well as control means of beyond design basic accident for the execute the main safety functions Category 4 includes PIE of severe accident (PIE SA), initiation of which in conditions of previous levels of defense-in- depth leads to transition in conditions level 4b of defense –in-depth , with of exceeding of the design limits for nuclear fuel damage in core and spent nuclear fuel in fuel pools , where control means for severe accidents execute the main safety functions.					Safety Guide or a different one.
Russia 3 comment 12		It shall be developed for the fulfillment of deterministic safety analysis: - complete lists of PIE get into every				X	See Resolution to “Russia 3, comment 11”

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		above mentioned category of PIE; - safety systems intended for execution of main safety functions shall be determined for every PIE taking into account the needs achievement for design target safety factors or safety criterions 4 - scenarios for implementation of deterministic safety analysis must be developed; - design analysis for identification of acceptance criterions must be developed in accordance with every scenario?					
Russia 3 comment 13		Scenarios for deterministic safety analysis implementation at every level of defense-in-depth have to be formed, as a rule, without regard to action of safety systems used to perform of similar functions at the next levels of defense- in-depth. In particular: - Scenarios for deterministic safety analysis implementation for PIE to category 1(PIE AOO – Anticipated operational occurrences) should be formed taking into account the action of special safety systems for normal operation which applied at the level 2 of defense-in-depth but without regard for actions of basic safety systems (BSS) used at level 3of defense-in-depth . Conditions for possible use at level 2				X	See Resolution to "Russia 3, comment 11". Systems availability for each plant state is treated in Section 7 of this Safety Guide.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		of basic safety system belong to level 3 defense-in-depth , as well as efficient safety systems belong to level1 defense-in-depth in some scenarios with PIE AOO should be brought into Guide for deterministic safety analysis; Scenarios for deterministic safety analysis implementation for PIE to category 2 basic design accident (PIE BDA) should be formed taking into account the action of basic safety systems which applied at the level 3 of defense-in-depth but without regard for actions of additional safety systems as well as control means for design accident (DEC) which are used at level 4a defense-in-depth . Conditions for possible use at level 3 of additional safety system belong to level 4 defense-in-depth , as well as efficient safety systems belong to level 1&2 defense-in-depth in some scenarios with PIE BDA at level 3 of defense-in-depth should be brought into Guide for deterministic safety analysis.					
Russia 3 comment 14		It is necessary to reveal the differences among eligibility criteria, targets deterministic safety analysis and safety criteria (2.5).				X	No specific proposal is provided and the text used para. 2.5 seem clear enough.
Russia 3 comment 15		It is necessary to reproduce which categories of PIE should use a				X	This aspect is treated in Section 7 of this Safety

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		conservative or realistic analysis.					Guide for each plant state (see "Analysis assumptions").
Russia 3 comment 16			"…Safety features for design extension conditions should not be credited in the analysis." However, in what follow, the term "safety features for design extension conditions" is not mentioned within description deterministic safety analysis for design extension conditions.			X	Paragraphs 7.47 to 7.51 indicate which systems can be credited in DEC without significant fuel degradation.
Russia 3 comment 17		It is recommended to revise the Guide for deterministic safety analysis taking into account the comments above.		n/a	n/a	n/a	n/a
Section 4							
EC-JRC – 9 (Observer/ unsolicited)	4/Title	SAFETY ACCEPTANCE CRITERIA	Chapter 4 focuses only on the third of the acceptance criteria category as identified in corresponding subsection 2 (which by the way takes exactly the same title).		*As indicated in the Resolution to "Turkey 24" about para. 2.6, the following change has been made there taking into account this comment:* "2.6. In this Safety Guide only the safety acceptance criteria that are …"	X	The existing title seems more adequate for this Section. However, a change has been incorporated in para. 2.6
France 11	4.3	4.3. Acceptance criteria should be	"Limiting" is not sufficient		*Paragraph 4.3 will*		*(Damage of some*

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		established for the entire range of operational states and accident conditions. These criteria should aim at ~~limiting~~ preventing damage to barriers against the release of radioactive material in order to prevent unacceptable radiological releases. Selection …"	here.		*be modified as follows:* 4.3. Acceptance criteria should … These criteria should aim at ~~limiting~~ preventing damage to relevant barriers against the release …"		*barriers is accepted in some accidents)*
Turkey 45	4.3 Line 3	"…material in order to prevent unacceptable radiological consequences ~~releases~~. Selection of the criteria should ensure…"	Better expression.		*Additionally to the change indicated in "France 11", para. 4.3 will also be modified as follows:* "…material in order to prevent unacceptable radiological ~~consequences~~ releases (thus also the consequences). Selection of the criteria should …"		*(Although the final objective is to prevent unacceptable consequences, this Safety Guide is dealing only with phenomena up to the releases)*
UK 17	4.5	"…subsequently approved by the regulatory body if so required by the regulatory for use in the …"	In the UK the regulator would not approve such criteria.		*Paragraph 4.5 will be modified as follows:* "…They are defined by regulatory requirements~~,~~ or proposed by the designer subject to regulatory acceptance ~~and subsequently approved by the regulatory body~~ for		*(The regulatory body should at least accept the criteria associated with integrity of barriers for being used in the Safety Analysis Report)*

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					use in the safety demonstration".		
IAEA/WES-17 (Unsolicited)	4.8	"4.8 Radiological acceptance criteria for normal operation should be typically expressed as effective dose limits for the plant staff and for the members of the public in the plant surroundings, …"	Ensuring consistency with the terminology used in Paras 4.6, 4.7, 4.9–4.11.	X			
Canada 35	4.9	*Suggest to re-word to* "4.9. The radiological acceptance criteria for anticipated operational occurrences ~~are typically comparable with annual dose limits for normal operation. They~~ should be more restrictive than for design basis accidents since their frequencies are higher."	The original sentence can be interpreted that the dose limit for AOOs needs to be accounted for in normal operation.	X			
Canada 36	4.12 Line 4	"…~~Examples of surrogate variables are: peak cladding temperature, departure from nucleate boiling ratio or fuel pellet enthalpy rise.~~	Adequate examples are provided in 4.13. Those in 4.12 can be deleted.	X			
Russia 1 Comment 14	4.12.	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 a barrier. 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 integrity of the barrier but cannot be measured directly. Examples of surrogate variables are: peak…	"surrogate variable" – is not a technical definition		*Additionally to the changes indicated in Canada 36, para. 4.12 will be modified as follows:* "… It is a common engineering practice to make use of surrogate variables(*) to establish an acceptance criterion or a combination of criteria for ~~that, if not~~		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					~~exceeded, will~~ ensur~~e~~ing the integrity of the barrier….”. *Footnote (*):* In this Safety Guide, the use of surrogate variables refers to the use of variables providing an indirect measure of another variable which direct measure is not possible		
Canada 37	4.13 Bullet 2	“…Criteria related to integrity of fuel cladding: minimum ~~nucleate~~ departure from ~~nuclear~~ nucleate boiling ratio, maximum cladding temperature, maximum local cladding oxidation;	Bullet contains errors. Correct term is “*minimum departure from nucleate boiling ratio*”.	1:X 2:X	D		
Finland 3	~~4.23~~ 4.13 Bullet 2	“… minimum ~~nucleate~~ departure from ~~nuclear~~ nucleate boiling ratio…”	correction of wording	X			
Turkey 46	4.13, L6 Bullet 2	“ … minimum ~~nucleate~~ departure from ~~nuclear~~ nucleate boiling ratio, …”	Typo.	X			
France 12	4.13 At the end of bullet 7	• “Criteria related to integrity of the containment and limitation of releases …of systems, maximum temperature in the containment. Criteria related to the integrity of the containment should be as close as possible to the pressure applied during the containment periodic tests.”	It makes no sense to use criteria related to integrity of the containment that is far from values tested during the periodic tests of the containment integrity. For example, a containment fragility curve could give a criteria of 9 bar for containment integrity pressure whereas the containment integrity			X	Containment periodic tests are not necessarily aimed at verification of the integrity of the containment. It is also justified by the comment on containment fragility curve, which should exist, but it is not used for periodic tests.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			pressure is tested only up to 5 bar during periodic tests…				
India 10	4.13 New Bullet	New bullet may be added: • Criteria for integrity of Reactor component like End shield, and CV	Design limits of pressure, temperature and minimum water level from shielding point of view may be considered.		*A new bullet will be added at the end:* • "Criteria related to integrity of any other component needed to limit radiation exposure, such as end shield in PHWR reactors: pressure, temperature and heat-up rate"		*(The resolution looks for consistency with other bullets; CV is already covered under containment)*
IAEA/WES-18 (Unsolicited)	4.15 Line 5	"…For design basis accidents, and for design extension conditions without significant fuel degradation, barriers to the release of radioactive material from the plant should maintain their integrity to the extent required …"	Editorial. The recommendation is valid for all DBAs.	X			
Russia 1 Comment 5	4.17	4.17. Although the assessment of engineering aspects important to safety may not be explicitly addressed in the safety analysis, it constitutes a relevant part of the safety assessment. Safety margins applied to the SSCs design should be commensurate with the probability of the loads they have to bear.	There is no disclosing of abbreviation SSC in the document.	X	*See "Russia 1 comment 4".* *All abbreviations will be expanded before publication.*		
Section 5							

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
France 13	~~5.5~~ 5.3	"(a) Identifying the important phenomena in the supporting experimental data and expected plant behavior and making sure that scale effects are properly taken into account"	Important differences between phenomena can result from scaling effects between experiments (usually small scale) and plant behavior (large scale)			X	Para 5.3 provides general steps for uncertainty quantification of the code. The guidance on scaling effect is provided in paras 5.29 to 5.31.
IAEA/WES-19 (Unsolicited)	5.4 (a)	"(a) Experimental data for the significant phenomena modelled. This would typically include a comparison against 'separate effect~~s~~ test' (SET) and 'integral effect~~s~~ test' (IET) experiments[footnote]." *Please assign a footnote to the term 'experiments' at the end of bullet (a), with the following text of the footnote:* "SETs are experimental tests which are intended to investigate a single physical process either in the absence of other processes or in conditions which allow measurements of the effects of the process of interest. IETs are experimental tests which are intended to simulate the behavior of a complex system with all interactions among the various processes occurring in various components of the system."	That is what the abbreviations 'SET' and 'IET' stand for in this context; see also Para 5.25, items (2) and (3). A brief explanation of the terms SET and IET should be provided in a footnote. Reference: OECD NUCLEAR ENERGY AGENCY, Review of Uncertainty Methods for Computational Fluid Dynamics Application to Nuclear Reactor Thermal Hydraulics, Report No. NEA/CSNI/R(2016)4, NEA, Paris (2016).		*Item (a) from para. 5.4 will be modified as follows:* "(a) Experimental data for the significant phenomena modelled. This would typically include a comparison against 'separate effect~~s~~ test' (SET) and 'integral effect~~s~~ test' (IET), see para. 5.25 ~~experiments~~;"		*(A footnote seems not necessary because the SET and IET are explained in para 5.25).*
Switzerland 1	5.5	(a) The users have received adequate training and ~~that~~ they appropriately understand the methods used in the code,	To a) Only the code vendors can be assumed to understand the (source) code. The	X	*Items (a) and (b) from para. 5.5 will be modified as follows:* (a) The users have		*(Regarding (a), the users should understand not only methods but also models).*

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		(b) The users are sufficiently experienced in the use of the code and fully understand its uses and limitations for the applications case (e.g. LOCA),	licenses holders (plants) are only "users". With this requirement, the licenses holders cannot make own calculations/analyses and must mostly outsource it to the code vendor or a limited amount of numbers of third party's worldwide. This wouldn't support the own know-how and competence and would as consequence create problems for the regulator. We propose to clarify this point. To b) Since computer codes nowadays comprises a lot of models and applications, it only can be expected that the user fully understand its uses and limitations of the application case (e.g. LOCA-calculations or reactivity transients). We propose to clarify this in adding application case.		received adequate training and ~~that~~ they appropriately understand the models and methods used in the code, (b) The users are sufficiently experienced in the use of the code and appropriately ~~fully~~ understand its uses and limitations for the application case (e.g. LOCA);		
Canada 38	5.5	(b) The users (or their supervisors) are	Adequate supervision can		*See Switzerland 1*	X	For safety analysis, a

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
	Item (b)	sufficiently experienced in the use of the code and fully understand its uses and limitations,	substitute for experience. If only experience analysts are allowed to do analysis, there is no way to employ a new analyst.				code user should have adequate experience using the code. A new analyst should be experienced by training before being involved into analysis work.
Jordan 9	5.6 (a)	"…The nodalization and the plant models are qualified and provide a good representation of the behavior of the plant, the nodalization is qualified when: • It has a geometrical fidelity with the plant. • It reproduces the measured nominal steady state condition of the plant. • It shows a satisfactory behavior in time dependent conditions."	Because it is necessary to confirm that that plant model and nodalization are valid and qualified.		*Item (a) from para. 5.6 will be modified as follows:* (a) The nodalization (see para. 5.38) and the plant models provide a good representation of the behavior of the plant,"	X	Qualification of nodalization is treated in para. 5.38.
Canada 39	5.7-5.20	Consider removing these sections.	These sections belong to software quality assurance process and are not a part of the safety analysis (SA) reports. However, Sections 5.21-5.38 should remain since code validation has a direct impact on SA results.			X	Paragraphs 5.7–5.20 are related to verification (including management) of the code. Both verification and validation are needed for computer codes used for safety analysis (see GSR part 4 (Rev.1) Requirement 18).
Canada 40	5.8 Line 3	"… The procedures should address, as a minimum, development control, document control, configuration of the code and testing and corrective	Editorial. Add comma after "as a minimum".	X			

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		actions."					
Canada 41	5.9 Line 3	5.9. To minimize human errors in code development, only properly qualified or supervised personnel should be involved in the development, verification and validation of the code. Similarly, in user organizations, only suitably qualified or supervised personnel should use the code.	Adequate supervision can substitute for experience.			X	Personnel who use computer codes for safety analysis should be qualified. Supervision is not enough to be sure that the analysis is conducted correctly.
IAEA/WES-20 (Unsolicited)	5.11	"5.11. If some tasks of code development, verification or validation are delegated by the code user to an ~~outside~~ external organization, those tasks should be managed to ensure quality within the ~~outside~~ external organization. The user should review arrangements within the ~~outside~~ external organization and should audit their implementation."	More appropriate terminology. The same term is also used e.g. in GSR Part 2 "Leadership and Management for Safety", GSG-4 "Use of External Experts by the Regulatory Body", and DS472 "Organization, Management and Staffing of a Regulatory Body for Safety" (revision and combination of GS-G-1.1, GS-G-1.5 and GSG-4).		*This resolution covers also Canada 42.* *Paragraph 5.11 will be modified as follows:* "… verification or validation are delegated ~~by the code user~~ to an ~~outside~~ external organization, those tasks should be managed to ensure quality within the ~~outside~~ external organization. The user's organization should review arrangements within the external ~~outside~~ organization and …"		
Canada 42	5.11 Last sentence	"…The user's organization ~~user~~ should review arrangements within the outside organization and should audit their implementation."	This is not done by individual users, but by other departments within the user's organization (e.g., our supply chain).		*Treated with IAEA/WES 20.*		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Turkey 47	5.11 Line 3 and 4	"…organization. The user should review the information on arrangements within the outside organization and their implementation to make sure that they are consistent with §5.7-§5.10…."	Auditing the outside organization by the user is not feasible. Reviewing the *information* on outside organization is much more achievable.		*See IAEA/WES 20 and Canada 42.*	X	Auditing an external organization seems feasible and it is recommended in this Safety Guide
Canada 43	5.20 Line 1	5.20. ~~The~~ A complex code may contain the integration or coupling of simpler codes. In such cases, verification of the complex code should ensure that…	It seems odd that a code can contain other codes – recursion. It may be clearer to reword as shown.	X			
ENISS - 10 (Observer / uninvited)	5.22 Line 4	"…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, and there are some rather exotic non-severe accident cases where the experimental data is limited (e.g. local recriticality during BWR emergency boration)		*See Canada 44*	X	Relaxation of the scope of validation mainly due to the lack of experimental data seems not acceptable. If there are no test data relevant to some phenomena, implementation of additional testing and provision for insufficient validation seems necessary.
Canada 44	5.22 End of para	"…validation might be relaxed for codes used in severe accident analysis, taking into account the limited relevant experimental data, in which case, additional reliance will be place on verification (as describes in 5.14-5.20)."	Validation of severe accident analysis is sometimes not possible. Only verification can be performed.	X	*The end of para. 5.22 will be modified as follows:* "… experimental data, in which case~~,~~ additional reliance should be placed on verification (see paras 5.14-5.20)."		
Canada 45	5.24	5.24. For complex analysis, the validation should be performed in two	Allows for validation to occur without relying on			X	User is responsible of the independent assessment phase.

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		phases: 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~~, a third party or independent individual.	code user.				
France 14	5.25 Item (2)	"… Separate effect tests should ideally be performed at full scale. If not, appropriate attention shall be paid to possible scaling effects."	Separate effect tests are often performed in reduced scale facilities.		*A new sentence will be added at the end of item (2) from para. 5.25:* "… Separate effect tests should ideally be performed at full scale. If not, appropriate attention should be paid to possible scaling effects (see paras 5.29 to 5.31)"		Item (2) from para. 5.25 deals with SETs. Scaling effects should be examined not only for SETs but also for IETs and this is treated in paras 5.29 to 5.31.
ENISS – 11 (Observer / uninvited)	5.25	"(4) Nuclear power plant level tests and … qualifying the plant model. For (2), (3) and (4), 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."	For example, for neutronic codes, refined Monte-Carlo calculations are used to support the validation of codes less detailed.	X	*The following text will be added after (4):* "For (2), (3) and (4) above, in the absence of relevant experimental data it is possible to enhance confidence on results by means of code to code comparison or bounding engineering judgement, to cover deficiencies in the full validation."		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Jordan 10	5.25 Page 29	It is suggested to add a figure (e.g. see Figure 3) to illustrate the hierarchy of experiments in the validation matrix. *(Note: See Figure 3 at the end of this Table)*	To illustrate that the required number of tests would decrease as the hierarchy moves from the basic and separate effect tests up to integral effect tests and plant operational data.			X	The use of figures is not a frequent practice in the Safety Guides and in this case it seems not necessary. Hierarchy in para. 5.25 can be implied by the structure (consecutive numbering) of the experiments.
France 15	5.30 Last sentence	"… The effects of phenomena that are not correctly represented should be addressed in other ways~~.~~ with a sufficient level of conservatism."	It is important to take into account phenomena not correctly represented in a sufficiently conservative way	X	*The last sentence of 5.30 will be modified as follows:* "… The effects of phenomena that are not properly ~~correctly~~ represented should be addressed in other ways~~.~~ taking into account the applicable level of conservatism."		
Canada 46	5.37 Last 2 sentences	"…~~The procedures, code documentation and user guidelines should be carefully followed to limit such user effects. Procedures include issues such as the way to compile the input data set and the means of selecting the appropriate models in the code and general rules for preparing the nodalization~~."	It should be sufficient to say qualified users and processed should be used.			X	These two sentences provide guidance to limit user effects; it seems preferable to keep them.
Canada 47	5.38 Line 4	"…A qualified nodalization that has successfully achieved agreement with experimental results for a given scenario should be used as far as	"As far as possible" does not allow any compromise (on cost or complexity, for example). "As far as			X	Reasons such as cost or complexity would not justify the change of a qualified nodalization.

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		~~possible~~ practicable for the same scenario when performing an analysis for a nuclear power plant."	practicable" is probably better.				If simpler nodalization is used in actual plant application it should be qualified before.
Canada 48	To add an additional paragraph after 5.38	5.38A. Validation should be documented and, the following information provided: • the modeled phenomena covered by the validation exercise; • key parameters for which computer uncertainty is determined; • description of the experiment or data used, • description of the computer input and options used in the validation; • description of the validation results.	Documentation of code validation is essential for confirmation of uncertainties used in deterministic safety analysis (DSA). This information needs to be included in DSA report.		*Paragraph 5.40 will be completed as follows:* "5.40. Each computer code needs to be…user can use the code properly. Description of the experiment or the key data used, description of the computer options used in the validation and description of the validation results should be included. The documentation should be available to all users." *(See Canada 51)*		Part of the proposed guidance is covered in para. 5.40, Section on "Documentation of Computer Codes".
Canada 50	5.39 Line 5	"…The input data should be a compilation of information found in ~~as-built and~~ valid technical drawings, operating manuals, procedures, …	For pre-construction safety analysis, there is no "as-built" plant.	X			
Canada 51	5.40 Last line	"…The documentation should be available to all code users."	Suggest adding explicit guidance that the users have access to the code documentation.	X	*Covered in Canada 48*		
Canada 49	5.40-5.42	Consider removing the section on "Documentation of Computer Codes"	Documentation of the computer codes is a part of QA activity and does not			X	Since the subsection on "documentation of computer codes" relates

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			need to be discussed in the DSA guidance.				to aspects such as code verification and validation, user effects or independent verification, it seems preferable to keep it in the Safety Guide.
Section 6							
Canada 52	6.2 Line 1 and 2	6.2. Uncertainties in computational predictions should be taken into account either implicitly by bounding them using a conservative, combined or ~~even~~ best estimate approach, or explicitly using a best estimate approach with quantification of uncertainties. …	Editorial. Three changes.	X	D *See Jordan 11*		
Jordan 11	6.2 Line 2	It could be better to avoid using the term "best estimate approach" when we refer to option 4 in table 1. It is suggested to replace it by "realistic approach" as indicated in option 4 in table 1 or "best estimate analysis" as described in section 2.15	In order to avoid unnecessary confusion between option 4 and option 3 in table 1.		*A footnote will be incorporated to Option 4, (Realistic(*)), in Table 1:* (*) For simplicity in this Safety Guide the term "realistic approach" or "realistic analysis" is meant "Best Estimate without quantification of uncertainties." *Additionally, para. 6.2 will be modified as follows*:		

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					6.2. Uncertainties in computational predictions should be taken into account either implicitly by applicable approaches (see Table 1) ~~bounding them using conservative, combined or even best estimate approach~~, or explicitly using … smallest margins to acceptance criteria); ~~see Table 1.~~		
Jordan 12	6.5 Line 3	It could be better to avoid using the term "best estimate approach" when we refer to option 4 in table 1. It is suggested to replace it by "realistic approach" as indicated in option 4 in table 1 or "best estimate analysis" as described in section 2.15	In order to avoid unnecessary confusion between option 4 and option 3 in table 1.		*See Resolution to Jordan 11*	X	The use of the term "best estimate approach" is consistent with para. 5.27 from SSR-2/1 (Rev. 1).
EC-JRC - 10 (Observer / uninvited)	6.6 Line 2	"…ensured. It should also ~~then~~ be demonstrated"	As currently worded, it is like if barriers integrity mentioned in the previous paragraph, were limited to demonstrating avoidance of cliff-edge effects, when cliff-edge effects constitute only one among different kind of challenges to the		*The sentence will be modified as follows:* "…ensured. It should ~~then~~ be demonstrated by sensitivity…"		

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			containment (in the frame of severe accidents). For instance, slow containment overpressurization, e.g. by noncondensable gases generation and steam heating up, can certainly threat the containment even if not belonging to cliff-edge phenomena characterization.				
IAEA/WES-21 (Unsolicited)	6.7 Line 5	*3rd sentence:* "These analyses should be used … for demonstration that a realistic change of the parameters does not lead to cliff-edge~~s~~ effects."	Usage of a consistent terminology throughout this Safety Guide is strongly recommended.		*A term consistent with the Safety Glossary will be used:* "… not lead to cliff edge~~s~~ effects. " *(See also IAEA/WES-37 to para. 9.18)*		
France 16	6.7 Last sentence, last line	"…However, it should be taken into account that when sensitivity analyses are carried with one-at-a-time parameter changes, misleading information may be obtained due to possible compensating or cumulating effects when several parameters change simultaneously."	When several parameters change simultaneously, there can be compensating effects but also the opposite, that is to say "cumulative" effects	X			
Turkey 48	6.7 Line 5	"…to cliff edges. However, it should be taken into account that when sensitivity analyses are carried out with"	Typo.	X	"…to cliff edge~~s~~ effects. However, it should be taken into account that when sensitivity analyses are carried out		

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					with…”		
EC-JRC – 11 (Observer / uninvited)	6.7 Line 7	In order to avoid such misleading effect masking the influence of uncertainty plant parameter values when applying one-at-a-time techniques, global sensitivity techniques should be used”.	Independent on the global sensitivity technique (as explicitly mentioned in the previous version of the same comment), recommendations for using more powerful and realistic global sensitivity techniques should be mentioned to indeed avoid for significant misleading information derived from too simple one-at-a-time techniques.			X	Specific technics of sensitivity analysis are beyond the scope of this Safety Guide. Also, it seems not clear what is meant by “global sensitivity techniques” (maybe it is equivalent to assessment of uncertainties).
Canada 53	6.9 First para	6.9. Deterministic safety analysis needs to incorporate a degree of conservatism that is commensurate with the safety analysis objectives and is dependent on the event class. For conservative ~~deterministic safety~~ analysis of anticipated operational occurrences and design basis accidents (see §2.14), ~~in addition to~~ instead of the fully conservative approach, one of two following options, or a combination of both should be considered: either:	A certain degree of conservatism should be considered in DSA to off-set any uncertainties associated with initial and boundary conditions and modelling. So, it is desirable to add additional wording on needs for conservatism. BEAU type analysis is a replacement for LOE (fully conservative analysis). Using both approaches is redundant.		*Paragraph 6.2 will be modified as follows*: 6.9 Deterministic safety analysis should incorporate a degree of conservatism which is commensurate with the safety analysis objectives and is dependent on the plant state. For conservative ~~deterministic safety~~ analysis of anticipated operational occurrences and design basis		

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					accidents (see §2.14), ~~in addition to~~ instead of the fully conservative approach, one of the two following options, or a combination of both, should be considered…”		
Turkey 49	6.11 Line 1	6.11. The selection of initial and boundary conditions should take into account of geometric changes, fuel	Better expression.			X	Unnecessary
Turkey 50	6.13, Line 7, bullet 2	“• Activity in reactor coolant system, including release of volatile fission products prior **to** or…”	Typo.	X			
Turkey 51	6.13, Line 18, bullet 10	“• Chemical and physical forms of radioactive substances release~~s~~d, in particular iodine”	Typo.	X	Chemical and physical forms of release~~s~~d radioactive material ~~substances~~, in particular iodine;		
Russia 1 Comment 2	6.13 Last bullet	A parameter characterizing the energy of a radioactive release (RR) should be added to para.6.13	Energy of RR affects its further propagation in the environment		*Last bullet will be modified as follows:* “Effective elevation ~~Height~~ of release to the environment taking into account the energy of the releases.”		
Russia 3 Comment 8	6.14 Line 3	It is offered to elaborate, what kinds of analysis “should consider a combination of validation of the code, use of conservatisms and use of sensitivity studies”.	”…To take into account uncertainties related to code models, the complete analysis should consider a combination of validation of the code, use of			X	No specific proposal is made and the existing formulation seems clear.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			conservatisms and use of sensitivity studies… /				
EC-JRC - 12 (Observer / uninvited)	6.16 Bullet 4	*[Note TO: To replace the existing 4th bullet by the following:]* • "~~There is a high risk of user effect i~~If conservative values are selected based on engineering judgment, there is a high risk that such selection implemented by the user, e.g. user effects, is not comprehensive and does not lead to conservative results.**"**	The current formulation seems to implicitly assume that user effects are intrinsically negative: "there is a high risk of user effects". User effects, even if lacking of a definition along the text or a supporting reference, should be taken just as the subjective effect that plays a significant role, whether **positive or negative**, when using computer codes.		*Last bullet of para. 6.16 will be modified as follows:* "~~There is a high risk of user effect i~~If conservative values are selected based on engineering judgment, there is a high risk that such selection implemented by the user is not appropriate and does not lead to conservative results."		
Armenia 4	6.19 whole	*We suggest to remove the first sentence of the paragraph (~~Initial condition that cannot occur at the same time in combination need not to be considered. For example, the limiting decay heat and the limiting peaking factors cannot physically occur at the same time of the fuel campaign)~~*	With regard to the current practice, the existing sentence proposes significant decrease of conservatism in cases of accident analysis, particularly, in cases of LOCA analysis.			X	The target of this para. is to discard the combination of conditions physically impossible to occur in the same moment. If such combination is used, it is for simplification and not because it is required. The example seems clear enough.
EC-JRC - 13 (Observer / uninvited)	6.20 Line 1	"…very limited time period and ~~therefore~~ with negligible frequency of occurrence…"	Very limited time of period and negligible frequency should not be taken as synonyms: certain Plant Operational States	X			

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			(POS) feature relevant contributions to overall risk. Therefore, it should be better to treat time period and low frequency as two independent matters to avoid neglecting a priori short transitional POSs.				
EC-JRC - 14 (Observer / uninvited)	6.21 Line 5	"…will be bounded by the results of best-estimate calculation plus uncertainty…"	'Calculation' should refer to the computer code reference simulation value which can also be named as best-estimate (since generated by a best-estimate computer code). 'Uncertainty', as referred in 6.21, also implicitly addresses uncertainty calculation.		*Paragraph 6.21 will be modified as follows*: "6.21. Uncertainties in DSA, in particular for AOOs and DBAs, may be addressed by the use of a best estimate computer code ~~in combination with best estimate~~ taking into account uncertainties in models, initial and boundary conditions and other input parameters. To ~~achieve~~ obtain conservative results of safety analysis, the effects of uncertainties on the results should be identified and assessed to confirm that the actual plant parameters will be		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					bounded by the upper and lower limits of the results of calculations ~~plus uncertainty~~ with an adequate confidence."		
Canada 54	6.23	6.23. 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.	Is it correct to say that an assessment of uncertainties is needed for identification of aleatory and epistemic sources? Surely identification would be needed first? I am not sure what was intended so I cannot propose a change.		*Paragraph 6.23 will be modified as follows*: "…, especially for the identification and separation of aleatory and epistemic sources of uncertainties. …"		
IAEA/WES-22 (Unsolicited)	6.24	"6.24 Quantification of uncertainties should be based on statistically combined uncertainties in plant conditions and code models to ensure with a specified probability, that a sufficiently large number of calculated results meet the acceptance criteria (see ~~§2.7~~ §2.5)."	Wrong reference is provided in brackets. Acceptance criteria are discussed in Para 2.5 while Para 2.7 does address methods for performing uncertainty analysis.		*Paragraph 6.24 will be modified as follows*: "… statistically combined uncertainties in plant conditions and code models (see para. 2.7) to ensure with…meet the acceptance criteria ~~(see para 2.7)~~. For analysis of AOO and DBA …"		
Section 7							
EC-JRC - 15	7	**"…** FOR DIFFERENT ~~PLANT~~	According to the IAEA		*The title will be*		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
(Observer / uninvited)	Title	OPERATIONAL STATES AND…"	glossary, plant states comprises operational states plus accident conditions		*modified as follows:* DETERMINISTIC SAFETY ANALYSIS FOR DIFFERENT PLANT STATES ~~AND ACCIDENT SCENARIOS~~		
Russia 1 Comment 15	7.3.	7.3. Decisions on the level of conservatism in performing deterministic safety analysis should include the following sets of input data or assumptions on: 1) Code models; 2) Plant operating parameters; 3) Control and limitation systems; 4) Active safety systems; 5) Passive safety systems; 6) ~~Safety features for~~ consequences of design extension ~~conditions~~ accidents; 7) Operator actions.	There are no safety features for the design extension accidents. Due to that reason these accidents are called "design extension".			X	See SSR2/1 (Rev. 1) para. 5.27: "This might require additional safety features for design extension conditions". *See also the Resolution to "Russia 1 Comment 11"*
IAEA/WES-23 (Unsolicited)	7.4	"Separate analyses of the source term should be carried out for each type of failures for which the phenomena that would affect the source term would be different. Typical kinds of accidents include: (a) loss of coolant accident with release of reactor coolant and fission products from the core to the containment~~,~~; (b) accidents by-passing the containment; ~~or~~ and (c) accidents taking place outside the containment, such as accidents in the spent fuel	With a view to support structuring and to improve the readability of the entire sentence, please include consecutive numbering of the various kinds of accidents for which separate analyses of the source term should be carried out.			X	The change suggested could let the reader think that the list is exhaustive whereas these are just examples. *See also IAEA/WES-7*

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		pool, accidents during manipulations with the irradiated fuel, or releases from the systems for treatment and storage of gaseous and liquid radioactive waste."					
Canada 55	7.6	7.6. Deterministic analyses of normal operation ~~should~~ can use an iterative process to support development of operational limits…"	Iterative process is not required for existing NPPs			X	The term "should" seems adequate given the scope of this Safety Guide
Pakistan 3	7.8	7.8. All 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, mid-loop operation, handling of irradiated fuel and off-loading of irradiated fuel from the reactor to the spent fuel pool	After shutdown and cooling, mid-loop operation have high risk of core damage and therefore, this needs to be considered.	X	*Additionally to the change proposed, the first sentence will be modified as follows:* "7.8 All ~~operating~~ modes of normal operation and relevant plant configuration covered by operational limits and conditions …"		
Armenia 3	7.8 Line 3	7.8. All operating modes of normal operation covered by operational limits and conditions should be analyzed, with a particular attention paid to transient operational regimes such as changes in reactor power, reactor shutdown from power operation, reactor startup, reactor cooling down, handling of fresh and irradiated fuel and off-loading of irradiated fuel from the reactor to the spent fuel pool and loading of fresh	Reactor startup is one of the important normal operation regimes leading to positive reactivity insertion in the core. Handling fresh fuel is among other important normal operative actions bearing the risk of criticality.	X	*The third/last request will be modified as follows:* "… to the spent fuel pool and loading of fuel into the core."		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		fuel to the core.					
IAEA/WES-24 (Unsolicited)	7.9	7.9. The safety analysis for normal operation should also include an analysis of the radiological situation in the plant and an estimate of the plant's releases of radioactive material to the environment."	Editorial.	X			
IAEA/WES-25 (Unsolicited)	7.10 1 sentence:	7.10. The analysis should assess whether normal operation of the plant can be carried out in such a way that plant parameter values do not exceed operational limits and conditions."	Usage of a consistent terminology throughout this Safety Guide is strongly recommended.	X			
IAEA/WES-26 (Unsolicited)	7.11	"The safety analysis for normal operation should include an analysis of the overall design and operation of the plant, in order to: (a) predict the radiation doses likely to be received by workers and members of the public; (b) assess that these doses are within acceptable limits; and (c) ensure that the principle that these doses should be as low as reasonably achievable has been satisfied. However, compliance with the radiological acceptance criteria ~~[3]~~ is not covered by this Safety Guide."	1st sentence: With a view to support structuring and to improve the readability of the entire sentence, please include consecutive numbering of the objectives for an analysis of the overall design and operation of the plant. 2nd sentence: The reference to the IAEA Safety Glossary (2016 Revision) is misleading and should be removed.		1st sentence: *The term "in order" will not be added. Added a consecutive numbering ((a), (b), (c), ...).* 2nd sentence: *The references will be updated in all the Safety Guide. In this case, the former [3] will be replaced by [4] and [5].*		
IAEA/WES-27 (Unsolicited)	7.15 Line 1 and 2	7.15. The initial conditions considered should be representative of all expected plant authorized modes, according to operational limits and	Usage of a consistent terminology throughout this Safety Guide is strongly recommended.	X			

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		conditions."					
Canada 56	7.17 Last line	"…The realistic analyses should aim at providing ~~the most possible~~ **a** realistic response of the plant to the initiating event."	Presumably this was intended to read "*aimed at providing the most realistic possible response*". However, this is too burdensome. "Most realistic possible" does not allow any compromise for cost or effort.	X			
Russia 2 Comment 10	7.18, third sentence	~~"…The anticipated operational occurrences category should include all the postulated initiating events which might be expected to occur during the lifetime of the plant."~~	To exclude this sentence as this subject is considered in section 3 of the present guidance in detail/			X	PIEs are treated in Section 3, though it seems meaningful to recall here that the plausible events should be handled without requiring safety systems to be triggered
Canada 57	7.19	7.19. Ideally, anticipated operational occurrences do not lead to any unnecessary challenges to safety equipment. ~~In addition, the anticipated operational occurrences should not lead to any unnecessary challenges to safety equipment primarily designed for protection in the event of design basis accidents.~~ It is therefore advisable to demonstrate by the analysis that, in case of the operation of plant control and limitation systems as intended, these systems will be capable of preventing the initiation of the safety systems. It is recognized that	Draft wording was too demanding. In most NPP designs there are some AOOs where safety system action is almost unavoidable.		*Paragraph 7.19 will be modified as follows:* 7.19. Typically, ~~In addition, the~~ AOOs should not lead to any unnecessary challenge~~s~~ to safety equipment primarily …of preventing the initiation of the safety systems. However, it is recognized that some AOOs require the actuation of safety systems.		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		some anticipated operational occurrences require actuation of safety systems.					
Canada 58	7.22 Line 1	"…requirement (for light water ~~cooled~~ reactors) that there…"	No dryout is identified as a typical AOO criterion for water cooled reactors. This criterion may not be met in CANDU loss of flow events.	X			
IAEA/WES-28 (Unsolicited)	7.23 2nd sentence	"…The radiological acceptance criteria for doses and correspondingly for releases for each anticipated operational occurrence~~s~~ should be comparable with annual limits for normal operation …"	Grammar.	X			
ILO-1 (Observer / Unsolicited)	7.23		The text should also identify that there should be negligible radiological impact on site as well as beyond the immediate vicinity of the plant.			X	This aspect is outside the scope of this Safety Guide (see para. 1.12).
Turkey 52	7.26 Line 1	7.26. Realistic analysis of anticipated operational occurrences should be performed ~~analysed~~ with best estimate…	Better expression.	X			
IAEA/WES-29 (Unsolicited)	7.27	7.27. Realistic analysis for design basis accident~~s~~ is not permitted; one of the conservative methods (Options 1, 2 or 3 from Table 1) should be used."	Editorial. The recommendation is valid for all DBAs.	X			
Russia 2 Comment 11	7.27	7.27. ~~Realistic analysis for design basis accident is not permitted;~~	To change the first sentence and exclude words that realistic			X	Current formulation seems correct in the framework of the DSA.

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			analyses for design basis accidents are forbidden. This is a wrong statement, such analyses are necessary for definition of the operator's actions, in this connection we propose present comments it is offered to supplement section 7 with a special subsection, containing recommendations on implementation of realistic analyses for justification of the operator's actions.				
Turkey 53	7.29 Line 1	7.29. The safety analysis should establish the performance characteristics and set points of the safety systems, and	Better expression.	X	*Paragraph 7.29 will be modified as follows:* "7.29. The safety analysis should establish the performance characteristics and set points of the safety systems ~~design capabilities, safety system set points~~, and operating procedures …"		
Canada 59	7.30	**This is a major comment.** 7.30. For conservative analysis of anticipated operational occurrences the	This document is not very clear with respect to treatment of AOOs. It		*Paragraph 7.30 will be modified as follows:*		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		technical acceptance criteria related to fuel integrity and radiological acceptance criteria should be the same as presented above for ~~realistic analysis of anticipated operational occurrences~~ design basis accidents.	should be clarified that for AOOs using conservative assumptions should be considered a DBA and hence subject to DBA analysis rules AND acceptance / dose criteria. The current document does not provide sufficient detail to determine if we apply dose limits for AOOs when doing AOO analysis using conservative methods.		"7.30. For conservative analysis of AOO the …radiological acceptance criteria should, in principle, be the same as presented above for realistic analysis of AOOs."		
Canada 60	7.31 Last line	"… acceptable effective dose limits are typically in the order of few ~~mSv/a~~ mSv per event."	Dose limits are normally expressed as "per event" not "per year".	X			
ILO-2 (Observer / Unsolicited)	7.31		The text should also identify that there should be no or only minor radiological impact on site as well as beyond the immediate vicinity of the plant. Guidance on dose criteria should be included to align with the frequency ranges given in para 3.26. The ILO, within its remit to protect workers, offers to co-operate in the definition of relevant guidance.			X	As indicated in ILO-1, this aspect is outside the scope of this Safety Guide (see para. 1.12).
Canada 61	7.32	… in anticipated operational	Editorial. "Design basis	X			

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
	Last line	occurrences or design basis accidents, some or all of the barriers …	accident" should be plural.				
IAEA/WES-30 (Unsolicited)	7.32	"7.32. Specific decoupling criteria should be defined in order to prove … that, in anticipated operational occurrences or design basis accidents, some or all of the barriers are able to limit the radiological releases to the environment."	Editorial. The recommendation is valid for all DBAs.	X			
Turkey 4	7.32	*Please, add the description of the decoupling criteria.*	Clarification to the meaning and relevance of the specific decoupling criteria with the section. (Are the criteria used to decide whether the thermal hydraulic and neutronic calculations to be decoupled from the radiological calculations? Or are the criteria used to decide whether a subsystem and primary structure to be analysed seperately? Or does it have any other meaning?)		*First sentence of para. 7.32 will be modified as follows:* "7.32. Specific ~~decoupling~~ technical acceptance criteria should be defined …"		
Turkey 54	7.32 Line 3	"…basis accident, some or all of the barriers are able to limit the ~~radiological~~ releases of radioactive material/substances to the environment."	See Comment No: 3 *(?? - TO: it seems to refer to Turkey 20)*	X			
IAEA/WES-	7.33	"… Thus, an anticipated operational	Editorial.	X			

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
31 (Unsolicited)	Dash 1, line 3	occurrence by itself should not generate a design basis accident, and a design basis accident should not generate **a** design extension condition~~s~~;”					
Canada 63	7.33 Dash 2	• “~~There should be no consequential loss of function of the safety systems needed to mitigate the consequences of an accident~~. Safety systems needed to mitigate the consequences of an accident should be capable of doing so including situations involving a loss of function of the safety systems;”	The current wording is overly restrictive. There could be a loss of function but system is still able to mitigate the accident.		*Dash 2 from para. 7.33 will be modified as follows:* - There should be no consequential loss of the overall function of safety systems needed to mitigate the consequences of an accident, although a safety system may be partially affected by the PIE ;		
Jordan 13	7.33 Dash 4	In the 4th para, the part underlined here [“-The pressure in the reactor and main steam systems <u>should not exceed the relevant design limits for the existing plant conditions</u>…” is suggested to be changed to "… - The pressure in the reactor and main steam systems should not significantly (more than 10–15 %) exceed the design value…"	According to section 4.4, "anticipated operational occurrences should have acceptance criteria that are more restrictive than those for less frequent events such as design basis accidents". Therefore, acceptance criteria for design basis accidents should be the same or less restrictive than acceptance criteria for anticipated operational occurrences described in section 7.22			X	Current wording is quite open. Detailed overpressure protection criteria are country specific. It seems preferable not to use figures here.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			& 7.30.				
Canada 62	7.33 *(Dash 6)*	Delete "fuel rod integrity" or mark this as an example for LWR only.	An exemption for PHWR should be stated as fuel rod integrity is not required to maintain coolable geometry. On the other hand, loss of structural integrity in a LWR results in core melt.		*Dash 6 from para. 7.33 will be modified as follows:* « … a coolable geometry and the structural integrity of the fuel ~~rods~~ assemblies (LWRs) should be maintained;»		
Turkey 5	7.33 6 Bullet	"- In design basis accidents with fuel uncovering and heating up, a coolable geometry ~~and structural integrity of the fuel rods~~ should be maintained;"	In DBA, limited fuel cladding failure is allowed thus structural integrity of fuel rod might not be maintained. A geometry that allows for adequate cooling should be maintained with or without a limited fuel cladding failure.		*See Canada 62*		
Turkey 55	7.35 Dash 1	"…affected by the postulated initiating event and its consequences ~~of the postulated initiating event~~ can be…"	Better expression.		*Dash 1 from para. 7.35 will be modified as follows*: "Normal operation systems … beginning of the postulated initiating event and that are not affected by the initiating event itself and by its ~~the~~ consequences ~~of the postulated initiating event~~ can		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					be assumed…"		
Jordan 14	7.35 Dash 1	The underlined phrase in this paragraph "-Normal operation systems that are in operation at the beginning of the event……" is suggested to be changed to "- Normal operation systems excluding control and limitation systems that are in operation at the beginning of the event……"	Because some of the normal operating systems such as chemical and volume control systems and protective protection systems have a positive impact on the course of accident. Therefore, these systems shouldn't be credited in the analysis.		*See Turkey 55*	X	In bullet it is indicated: « … No credit should be taken for the operation of the control systems in mitigating the effects of the initiating event ; ». This seems clear enough.
Canada 64	7.35 Dash 3	- Safety systems designed and maintained as safety grade (in accordance with the rules for quality assurance, periodic testing, use of accepted design codes and equipment qualification) should be assumed to operate with conservative performance~~s~~;	Editorial. "Performances" should be singular.	X			
Japan 2	7.35. Dash 4 (line 16)	Please delete the lase sentence of the ~~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;~~	This example is not a common practice in the States. The "one rod stuck margin" is the design requirement for reactor shutdown system and is not regarded as the single failure assumed in safety analysis in Japan.	X	*See Canada 65*		
Canada 65	7.35 Dash 4	"- In accordance with the single failure criterion, a ~~A~~ single component failure should be assumed to occur in the	Reword for clarity. The requirement to assume	X	*See Japan 2*		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		operation of the safety groups required for the initiating event, in addition to the initiating failure and any consequential failures ~~(the Single Failure Criterion)~~. Depending 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~~;	failure of the insertion of the control rod that has the greatest effect on reactivity is pre-judging the outcome of the Single Failure Criterion. Recommended that discussion about control rod is removed.				
France 17	7.35 Dash 5	"- Safety features designed specifically for design extension conditions should not be credited in the analysis."	The initial text is unclear (for example the containment is used for design extension condition …)	X	*Dash 5 will be modified as follows:* "Safety features specifically designed for design extension conditions …"		
Jordan 15	7.35 Dash 5 Page 44	It is suggested to add to the 5th dash: "… - Safety features for design extension conditions should not be credited in the analysis in order to comply with the independence principle between the levels of defense in depth"	In order to provide justification to explain why Safety features for design extension conditions should not be credited in the analysis.		*See France 17*	X	It seems better not to refer to DiD in a bullet devoted to systems crediting
EC-JRC - 16 (Observer / uninvited)	7.35 Dash 5 Line 18	"- Dedicated safety features for …"	DEC scenarios can be mitigated by using standard safety systems initially designed to cope with DBAs. By stating 'dedicated' such mitigating			X	*See France 17*

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			systems not taken into account in DBA analysis are limited to post-Fukushima devices (non-permanent equipment, additional systems designed to perform only in DEC PIEs, etc.).				
Armenia 5	7.35 whole	*We suggest to remove first bullet of this paragraph:* 7.35. The conservative considerations regarding the availability of plant systems should typically include the following: ~~- 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 postulated initiating event can be assumed to continue to operate;~~	Taking into account the current practice, the existing sentence proposes an optimistic assumption with regard to availability of the systems in case of DBA, so it may decrease conservatism in DBA analysis.			X	*See Jordan 15.* It is the usual practice to credit operation continuity when relevant, e.g. regarding main coolant pumps. Otherwise fuel integrity criteria could generally not be met in DBC2 (see Table 2)
Canada 66	7.36 Line 1	7.36. If maintenance is allowed, the unavailability of the concerned train of the safety system should be taken into account.	Editorial. Add comma after "allowed".	X			
Turkey 56	7.37 Line 2	"…and ~~starting the~~ initiation of necessary actions. The corresponding timing claimed should be justified and validated for…"	Better expression.	X	*First sentence from para. 7.37 will be modified as follows:* "… diagnosis of the event and for initiating ~~starting~~ the necessary actions. The corresponding …"		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Canada 67	7.37	**This is a major comment.** 7.37. For conservative safety analysis, credit should not be taken for operator diagnosis of the event and starting the actions until after a conservative delay time. The corresponding timing claimed should be justified and validated for specific reactor design; for example earlier than in 15 minutes if performed in the control room, or 30 minutes for ~~the~~ field actions	Given that control room staff are trained to respond to the full spectrum of events, the suggested times in the draft are unreasonably long. Suggest using the standard 15 and 30 minute times as used in current analyses. Alternatively, it should be made clear that these times apply only for new NPPs (though even for them, they appear to be unreasonably long). Editorial: First sentence is incomplete. Editorial error on last line. Delete "the"		*See Turkey 56*	X	*Scope of this Safety Guide (SG) is indicated in para. 1.6:* "This SG focuses primarily on the DSA for the design safety of new NPPs and, as far as reasonably practicable or achievable, also the safety re-evaluation or assessment of existing NPPs…
Japan 3	7.37 Line 2	"… the actions. The corresponding timing claimed* should be justified and validated for specific reactor design; for example earlier …" (*) Current practices in some States are that ~~for example~~ earlier than in 30 minutes if performed in the control room, or 60 minutes for the field actions.	The 30 minutes and 60 minutes rules should be stated in a footnote as some States practices.			X	The change seems not necessary; there are many examples in the Safety Guide
Turkey 57	7.38 Line 1	7.38. The actions of the plant staff to prevent or mitigate the accident ~~by taking correct actions~~ should…	Better expression.			X	It seems better to refer to correct actions in the Safety Guide
Switzerland	7.38	"... working environment in the control		n/a	"ample" is the right	n/a	

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
2	Line 4	places, a mple information, written procedures and training."	The red marked text seems to be a typo.		term to be used and it seems that it was properly written in the draft.		
Turkey 58	7.40 Line 2	"…design basis accidents should take into account of uncertainties in the initial conditions and boundary	Better expression.			X	"Take account of" seems correct
Turkey 59	7.40 Line 4	"…Section 6 should be applied in full for these categories of plant states. The aim is to ensure ~~there is~~ with high **…"**	Better expression.	X			
Japan 4	7.42. Line 4	The last sentence is unclear. Please replace "Otherwise" by clear wording. ~~"Otherwise~~ If a best estimate plus uncertainty methodology is applied, uncertainties on safety systems performances are included in the overall uncertainty analysis."	Clarification.	X			
IAEA/WES-32 (Unsolicited)	7.43	7.43. In addition to the postulated initiating event itself, a loss of off-site power (LOOP) may be considered as additional conservative …"	Please introduce abbreviations before using them in a Safety Standard.	X			
Russia 1 Comment 6	7.43	7.43. In addition to the postulated initiating event itself, a loss of off-site power may be considered as additional conservative assumption. If LOOP is considered…	There is no disclosing of abbreviation LOOP in the document.	X	*See IAEA/WES 32*		
Canada 68	7.44 Whole para.	7.44. In line with the general rules for deterministic safety analysis, the source term evaluation of anticipated operational occurrences and design basis accidents would consist in taking into account all significant physical	This paragraph is unclear and the sentence is too long. Suggest it is revised as shown. The text in brackets is redundant and should be	X			

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		processes occurring during an accident and ~~introducing to the modelling the~~ using conservatively determined numerical values of initial data and coefficients ~~(which reflects the conservative approach)~~ on a plant specific basis.	deleted. The whole subsection is about the conservative approach.				
Canada 69	7.45	[Replace 7.45 with classification criteria that do not depend on outcome of the analysis results]	Event categorization of Design Extension Conditions into those with or without significant fuel failures does not make sense. This means classifying events based on the outcome, which is clearly the wrong thing to do.			X	Plant states are defined in SSR-2/1 (Rev.1), including DEC without significant fuel degradation, and the guidance provided in this Safety Guide has to be adapted to them.
Canada 70	7.49	7.49. According to the independence principle between the levels of defence in depth the normal operation systems including control and limitation systems should not be credited in the short-term analysis of design extension conditions …"	It is not clear what this clause is intended for. Normal systems and equipment is credited under DEC.		*(New 7.50; see Jordan 16)*	X	In new plants, normal operation systems should not be credited in DEC. According to para 1.6, this Safety Guide primarily focuses DSA for new plants.
Canada 71	7.50 Line 2	7.50. Non-permanent equipment should not be considered in the short term for demonstration of adequacy of the nuclear power plant design.	First sentence says that non-permanent equipment should not be considered. Second says that it may be considered in the long term. The two sentences are contradictory. [Also 7.64.]		*(New 7.51; see Jordan 16)*	X	This para. was discussed and agreed, including the proposed change, in NUSSC-42. It seems better not to incorporate the change proposed.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Russia 1 Comment 7	7.50 and also 7.64	7.50. Non-permanent 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 accordance with the emergency operating procedures or accident management guidelines. The time claimed for availability of non-permanent equipment should be justified.	There is no definition of "non-permanent equipment" in the document.		*(New 7.51; see Jordan 16)*	X	The term "non-permanent equipment" is used in SSR2/1 (Rev. 1), e.g. in para. 6.28 (b)
EC-JRC - 17 (Observer / uninvited)	7.50 & 7.64	See comments	The same text is found in both paragraphs.		*(New 7.51; see Jordan 16)*	X	The same requirement applies to both paras. No change seems necessary.
EC-JRC - 18 (Observer / uninvited)	7.50	7.50. Alongside meeting with the aforementioned conditions, non-permanent equipment may be considered available only for design extension conditions provided they are stored onsite and evidence has been given to test their operability before the time they are needed according to the accident evolution simulation. Non-permanent equipment, if stored offsite, should not be considered for demonstration of adequacy of the nuclear power plant design …"	Current text neglects the possibility of using portable equipment to demonstrate an adequate plant response as a general rule (with the provided exceptions in the footnote after the first 8 hours). However, several portable equipment, or semi-portable equipment, might need to be quickly put in place for instance, to inject water into the reactor cavity to achieve IVMR. If done after 8 hours, the		*(New 7.51; see Jordan 16)*	X	*See Canada 71:* This para was discussed and agreed, including the proposed change, in NUSSC-42. It seems better not to incorporate the change proposed

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
			RPV will have probably failed. This action, taken as a mere instance, may be placed in the procedures right after transiting to SAMGs or even sooner. Therefore, I do not think it is consistent with post-stress tests backfitting measures and acceptance results by nuclear regulatory bodies to avoid using those systems as stored onsite (since offsite equipment is much more difficult to control and put quickly in place).				
Canada 72	7.51 Line 2	7.51. Best estimate assumptions should be used for the analysis of design extension conditions. Some conservative ~~Conservative~~ assumptions as described for design basis accidents may be used to the extent practicable.	First sentence says "best estimate assumptions should be used"; the second says "conservative assumptions may be used". The sentences are inconsistent.	X	*(New 7.52; see Jordan 16).* *See Turkey 60. Paragraph 7.51 (new 7.52) will be modified as follows:* 7.51. Best estimate assumptions ~~should~~ might be used regarding operator actions for the analysis of DECs. However, some ~~C~~conservative assumptions as described…"		

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Turkey 60	7.51 all	Paragraph 7.51, in its entirety, needs to be moved under the following heading (Analysis assumptions and treatment of uncertainties).	This text is not about Operator Actions, but rather about Analysis Assumptions. As such, it needs to be moved under the following heading.	X	*(New 7.52; see Jordan 16).* *See Canada 72*		
Turkey 61	7.51 all	7.51. Planned operator actions performed in accordance with operating procedures and accident management guidelines should be considered in the analysis.	Suggested text for Operator Actions.		*(New 7.52; see Jordan 16).*	X	*See resolution to Canada 72*
Canada 73	7.53 Line 4	However, in line with the general rules for analysis of design extension conditions, ~~the~~ best estimate analysis ~~not always~~ without requiring a quantification of uncertainties can be used, but see §7.54 and §7.67.	Revise for clarity.	X	*(New 7.54; see Jordan 16).*		
France 18	7.55	7.55. For design extension conditions without significant fuel degradation, single failure criterion does not need to be applied systematically. Equipment redundancy should be considered on case by case approach. Furthermore, unavailability of a system or component due to maintenance does not need to be considered, if the maintenance time is negligible.	For example, a redundant instrumentation may be requested. It is important to add that maintenance is not considered only if the maintenance time is small.		*(New **7.49**; see Jordan 16).* *This para. will be modified as follows:* 7.49~~55~~ For DECs without significant fuel degradation, the single failure criterion does not need to be applied. Furthermore, unavailability of safety features for this category of design extension	X (proposed change)	Redundancy can be implemented to improve the reliability claim but it is not required by the analysis rules. "Negligible time" for maintenance is over conservative for DEC.

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					conditions ~~a system or component~~ due to maintenance does not need to be considered." *(It covers Canada 74)*		
Jordan 16	7.55	It is suggested to move section 7.55 to be within the sections related to subtitle of **Availability of systems**	As it was done in the previous sections, application of single failure criterion was addressed within subtitle of **Availability of systems**.	X	*Former para 7.55 will be placed as 7.48A (new **7.49**). Numbering of paras from existing 7.49 to 7.54, will be updated.*		
Canada 74	7.55 Line 1	7.55. For design extension conditions without significant fuel degradation, the single failure criterion does not need to be applied.	Editorial. Add "the". [Also 7.63 for DEC with core melting.]	X	*(New 7.49; see Jordan 16).*		
UK 18	7.55 Line 2	*Change final sentence to* "…Furthermore, unavailability of a system or component due to maintenance may ~~does~~ not need to be considered"	In the UK our initial expectation would be that a diverse safety system would remain operable during a plant maintenance condition on a single train of the system since this is a planned state that will definitely be entered into. In exceptional circumstances it might be possible to argue that it is not reasonably practicable to ensure availability during the maintenance		*(New 7.49. See Jordan 16)*	X	Current formulation seems adequate. The aim of DEC-A analysis is not necessarily to cover any possible initial plant state as they are meant to prove that core melt frequency is acceptably low

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			condition but this would need to be demonstrated. For example it is quite easy to incorporate redundancy into a C&I system during the design phase for a new power plant.				
Canada 75	7.56 Bullet 2	- the plant structures, systems, and components (e.g., the containment design) and procedures are capable of preventing a large radioactive release or an early radioactive release , including containment by-pass;	Editorial. Delete space before comma where indicated.	X			
Finland 4	7.57.	7.57. The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design combined with implementation of severe accident procedures or ~~accident~~ management guidelines."	Add: severe accident procedures or There could be also severe accident procedures available.		*Paragraph 7.57 will be modified as follows:* "… combined with implementation of procedures or guidelines for accident management ~~guidelines~~."		
IAEA/WES-33 (Unsolicited)	7.57	7.57. The safety analysis of severe accidents should demonstrate that compliance with the acceptance criteria is achieved by features implemented in the design combined with implementation of a plant specific accident management programme (including emergency operating procedures and severe accident management guidelines; see	Ensuring consistency with the terminology, concepts and approaches established in the Draft Safety Guide DS483 "Severe Accident Management Programmes for Nuclear Power Plants", which provides recommendations for the development and			X	*See Finland 4.* No need to enter here into the suggested details. DS483 is referenced in footnote 3 (para.2.18)

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		[18]).”	implementation of a plant specific accident management programme (AMP) during all modes of operation for the reactor and the spent fuel pool. The AMP does include emergency operating procedures (for the preventive domain of accident management) and severe accident management guidelines (for the mitigatory domain of accident management). A new reference [18] to DS483 should be inserted in Para 7.57.				
Canada 76	7.59 Line 3	“…Examples of acceptance criteria for design extension conditions analysis would include limitation of the containment pressure**,** containment water level**,** temperature and flammable gases concentration and stabilization of molten corium….”	Water level in containment may need to be considered for some reactor designs.	X			
Canada 77	7.60. Line 1	7.60. On site radiological acceptance criteria should ensure habitability of the control locations (i.e. control room, supplementary control room, field control areas, other emergency response facilities) and the locations required to travel between control locations.	Personnel may need to access control areas outside of the control rooms (main and secondary) to perform field actions to mitigate or stop the event.		*First sentence from 7.60 will be modified as follows:* “… and other emergency response facilities and locations) and the areas used to move		

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Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
					between control locations. In particular…"		
IAEA/WES-34 (Unsolicited)	7.60	7.60. On site radiological acceptance criteria should ensure habitability of the control locations (i.e. control room, supplementary control room and other emergency response facilities). In particular, the radiation level (e.g. ambient equivalent dose rates, activity concentrations in the air, etc.) in the control room, the supplementary control room and ~~in~~ other emergency response facilities ~~of~~ on the site ~~(e.g. ambient equivalent dose rates, activity concentrations in the air, etc.)~~ should allow for adequate protection of their occupants, such as emergency work-ers, according to Requirement 11 and Requirement 24, §6.25 from GSR Part 7 [11]."	In addition to Req. 11, also Para 6.25 subordinated to Req. 24 of GSR Part 7 is relevant here; it states: "*For facilities in category I, emergency response facilities separate from the control room and supplementary control room shall be provided so that:* *(a) Technical support can be provided to the operating personnel in the control room in an emergency (from a technical support centre).* *(b) Operational control by personnel performing tasks at or near the facility can be maintained (from an operational support centre).* *(c) The on-site emergency response is managed (from an emergency centre).* *These emergency response*		*Second sentence of 7.60 will be modified as follows:* "… In particular, the radiation level (e.g. ambient equivalent dose rates and activity concentrations in the air) in the control locations ~~room and in other emergency response facilities~~ of the site ~~(e.g. ambient equivalent dose rates, activity concentrations in the air, etc.)~~ should allow for adequate protection of their occupants, such as emergency workers, according to ~~R~~requirements 11 and 24 from GSR Part 7 [xx].		

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			facilities shall operate as an integrated system in support of the emergency response, without conflicting with one another's functions, and shall provide reasonable assurance of being operable and habitable under a range of postulated hazardous conditions, including conditions not considered in the design."				
France 19	7.62	7.62. Consideration of availability of equipment, including the instrumentation to manage the accident and to inform authorities, credited to operate under severe accident conditions should include - Circumstances of the applicable initiating event, including those resulting from external hazards (e.g. station blackout, earthquakes) and - Environment (e.g. pressure, temperature, radiation) and time period for which the equipment is needed	It is important to point out that instrumentation is also part of equipment used under SA conditions. Instrumentation used to inform authorities is part of this instrumentation.			X	This paragraph should not have different purposes. Here it is dedicated to qualification, not to the identification of necessary functions.
France 20	7.63	7.63. For design extension conditions with core melting, single failure criterion does not need to be applied	For example, a redundant instrumentation may be requested.			X	Redundancy can be implemented to improve the reliability claim but it is not required by the

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		systematically. Equipment redundancy should be considered on case by case approach. Furthermore, unavailability of a system or component due to maintenance does not need to be considered, if the maintenance time is negligible.	It is important to explain that maintenance is not considered only if the maintenance time is small				analysis rules. "Negligible time" for maintenance is over conservative for DEC
Spain 2	7.64	We propose to clarity the term Non-permanent equipment. Function, necessary for,…	To limit the scope of interpretations			X	See "Russia 1, comment 7". The term "non-permanent equipment" is used in SSR2/1 (Rev. 1), e.g. in para. 6.28 (b).
France 21	7.64	7.64. Non-permanent equipment should not be considered for demonstration of adequacy of the nuclear power plant design without proper justification of its safety classification, its availability and of the feasibility of its implementation in due time under severe accident conditions[13]. Such equipment is typically considered to operate for long-term sequence and is considered available in accordance with the emergency operating procedures or accident management guidelines.~~. The time claimed for availability of non-permanent equipment should be justified13.~~	Non permanent equipment could be used IF time claimed for availability and feasibility is well justified.			X	See Canada 71 about 7.50. This para. was also discussed and agreed in NUSSC-42. It seems better not to incorporate the changes proposed
Russia 2 Comment 12	7.65	7.65 .The same operator's actions ~~should be considered as for design extension conditions without~~	To change the wording of this para and exclude the statement that the		*7.65 will be modified as follows:*		

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		~~significant fuel degradation (so §7.51)~~ are provided according to accident management guidance.	operator's actions at severe accidents remain the same as at design extension conditions without fuel melt. This is a wrong statement as such actions cannot be identical. Actions of the operator are determined by the accident management guidance.		7.65. The same assumptions regarding operator actions should be considered …"		
Russia 2 Comment 13	Section 7	DETERMINISTIC SAFETY ANALYSIS FOR JUSTIFICATION OF THE OPERATOR ACTIONS. Specific objectives of the analysis. Acceptance criteria. Availability of systems. Operator's actions. Analysis assumptions and treatment of uncertainties	To supplement section 7 with a special subsection, containing recommendations on implementation of realistic analyses for justification of the operator's actions. This subsection has to reflect the same issues, as other subsections of this section, namely: specific objectives of the analysis, acceptance criteria, availability of systems, operator's actions, analysis assumptions and treatment of uncertainties.			X	According to emergency operating procedures, relevance of operator actions has to be proved by DBC (design basis conditions; see Table 2) and DEC analysis, not by dedicated analysis. Additional realistic analysis may be performed but it is not part of the DSA.
ENISS - 12 (Observer / uninvited)	7.65	"7.65 The operator actions for analysis of design extension conditions with core melting should be considered as for analysis of design extension conditions without significant fuel degradation (see §7.51)."	The statement is not clear as it is written.		*See Russia 2 comment 12*		

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Canada 78	7.67 Line 2	7.67. Analysis of severe accidents should be performed using a realistic approach in Table 1~~, (~~to the extent ~~possible)~~ practicable.	"To the extent possible" does not allow any compromise for cost or effort. Change "possible" to "practicable". Editorial. Text is clearer without the brackets. Suggest they are removed.	X			
EC-JRC 19 (Observer / uninvited)	7.67 Line 2	"…extent possible). Even though ~~since~~ explicit quantification of uncertainties may be impractical due to complexity of the phenomena and insufficient experimental data, the magnitude of the uncertainty advises that a thorough analysis of the main sources of uncertainty consistent with the state of the art should be integrated in the calculations~~sensitivity analyses should be performed to demonstrate the robustness of the results and the conclusions of the severe accident analyses~~."	Nowadays significant efforts have been and are being performed to address uncertainties in the field of severe accidents (e.g. OECD/NEA STEM, USTA proposal to HORIZON2020, etc.). Rather than suggesting to perform some sensitivity analyses, the guide should recommend to take account of the accuracy of the results by performing and implementing an analysis of uncertainty, even if limited according to the state of the art.			X	The USTA proposal was made to NUGENIA in April 2016. It would not be appropriate to base guidance on unfinished work.
France 22	7.69	According to §2.1, the demonstration of 'practical elimination' of the possibility of certain conditions arising that could lead to a large radioactive release or an early radioactive release include deterministic considerations (in terms of design, fabrication,	Clarification		*7.69 will be modified as follows:* "… early radioactive release include deterministic considerations together with	X Last change proposed	See Resolution to Turkey 7 (General comments); the need of having an adequate safety classification and the corresponding guidance available

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		testing, inspection and evaluation of the operating experience) ~~supplemented by a number of investigations such as those related to fabrication, testing, inspection and evaluation of the operating experience and~~ supplemented by probabilistic considerations, taking into account the uncertainties due to the limited knowledge of some physical phenomena. Proper safety classification (high level) should be applied to SSCs used for practical elimination.	Do not forget to request high level safety classification to SSCs used for practical elimination		engineering aspects such as design, fabrication, testing, inspection and evaluation of the operating experience and supplemented ~~by a number of investigations such as those related to fabrication, testing, inspection and evaluation of the operating experience and~~ by probabilistic considerations, …"		(SSG-30) is indicated in para. 7.2.
Canada 79	7.69	IAEA should refer to IAEA TECDOC 1791 which was released earlier this year for further clarification with respect to " practically eliminated	Some guidelines are provided in other IAEA documents.			X	Reference to a Tecdoc is not used in a Safety Guide
France 23	7.70	7.70. Demonstration of 'practical elimination' of the possibility of certain conditions arising should include, where appropriate, the following steps: • Identification of undesired conditions (challenges) potentially endangering the containment integrity or by-passing the containment, resulting in an early radioactive release or a large radioactive release; • Challenges should be addressed; in case this is not possible, design and	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		D See EC-JRC-20: *2nd bullet will be modified as follows:* Challenges should be addressed by implementing ~~in case this is not possible,~~ design and operational provisions ~~should be implemented~~ in order to 'practically eliminate' the	X Bullet 3	It seems preferable not to remove but to modify bullet 3. The sensitivity studies are important; if "threshold values" are defined, sensitivity studies can be performed to validate them (threshold values).

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		operational provisions should be implemented in order to 'practically eliminate' the possibility of those conditions arising; ~~• 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.	designed such to guarantee those margins. Here the object of the sensitivity studies is not clear, so consider deletion		possibility of those conditions arising; *3rd bullet will be modified as follows:* "… margins exist to address uncertainties regarding the demonstration with high level confidence that the possibility of the referred conditions has been 'practically eliminated'~~and to avoid cliff-edge effects~~"		
EC-JRC - 20 (Observer / uninvited)	7.71 new	7.71.~~ Although probabilistic targets can be set, demonstration of the 'practical elimination' of certain event sequences arising that could 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 a justification for not providing reasonably practicable safety features.~~ 7.71. 'Practical elimination' is achieved if it would be physically impossible for the conditions to arise or if these conditions could be considered with a high level of confidence to be	The definition of 'practically eliminated' phenomena presented in SSR-2/1 and taken as reference, points at two different types of situations to occur: whether (i) they are physically impossible or (ii) they are unlikely to arise with a high level of confidence. Para 7.72 deals with the first of the two categories, e.g. 'physically impossible', whereas para 7.71 focuses on the probabilistic		*Paragraph 7.71 will be modified as follows:* 7.71.Although probabilistic targets can be set, demonstration of the 'practical elimination' of certain event sequences arising that could 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~~		

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		extremely unlikely to arise. With respect to the latter, early radioactive release or a large radioactive release should not be based solely on low probability numbers. Scenarios featuring such conditions should rather be deterministically imposed and their elimination should be based on high level performance of dedicated safety features which should make the conditions extremely unlikely to arise.	argument to consider the practically eliminated clause. Nonetheless, how can then be interpreted the probabilistic argument presented in the 'practically eliminated' definition, i.e. *unlikely to arise with a high level of confidence* on the light of para 7.71? Please take not that this condition is presented in SSR-2/1 as a different one with respect to the 'physically impossible' condition. For instance, could passive FCVS be taken to support the physically eliminated condition by the probabilistic argument of containment pressure by slow overpressurization caused by steam and non-condensable gases build-up, even if subject to fail (yet with a very low probability)? If not, which conditions can stand behind the probabilistic argument (*extremely unlikely to arise*)?		~~value cannot be considered a justification for not providing reasonably practicable safety features~~. Such event sequences should rather be deterministically defined and their 'practical elimination' based on the performance of safety features making the events sequences extremely unlikely to arise.		

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			The probabilistic argument should look not at the frequency of the sequence itself but rather, after deterministically imposing the related sequences, safety features high level of performance. This way we do not rely on neglecting low frequency scenarios but we deterministically impose them and afterwards neglect them because of relying on high confidence systems. Otherwise it is difficult to explain the probabilistic argument included in SSR-2/1.				
UK 19	7.71 - 7.72	*Section needs rewriting.*	This section on "practicably elimination" needs more work. Given that this is a new concept more guidance is needed on what "demonstrating something has been practicably eliminated" actually means in practice. It tends to define it by saying what it is not (i.e. it should not be based solely on low probability numbers). In particular,			X	No proposal of text is made.

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			the section on uncontrolled reactivity accidents appears to ignore heterogeneous boron dilution faults for which the negative reactivity coefficient will be too slow to protect against the fault.				
EC-JRC - 21 (Observer / uninvited)	7.72 Line 1	"…resulting in an early or large radioactive release…"	The 'physical impossible' condition clause should affect both types of severe accident category of scenarios to be 'practically eliminated', i.e. early but also large radioactive release type.		*See Resolution to France 24*		
France 24	7.72	"7.72. Where a claim is made that the conditions ~~potentially resulting in an early radioactive release~~ are 'physically impossible', it is necessary…"	On the basis of the definition given in the previous paragraphs, why the physical impossibility is limited here to "early radioactive releases" (isn't it "large early releases"?).		According to SSR2/1 (Rev.1), para. 5.27, this para. *7.72 will be modified as follows:* "7.72. Where a claim is made that the conditions potentially resulting in an early radioactive release or a large radioactive release are 'physically impossible'…"		
Section 8							
EC-JRC 22	8.3	"…provide a list of all ~~plant states~~	Editorial			X	Existing sentence is

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(Observer / uninvited)	Line 1	postulated initiating events considered…"					correct
UK 20	8.3		See comment UK-19.			X	As well as in UK-19, no specific proposal is made. *UK-19 deals with paras 7.71-72, suggesting rewriting the subsection on practical elimination.*
IAEA/WES-35 (Unsolicited)	8.3 1st sentence:	"8.3. The ~~S~~safety report should provide a list of all plant …"	Editorial.	X			
Turkey 62	8.5 Line 4	"…and validated by the user, or certified by an authority to that effect (see §5.14 to §5.38)."	In some Member States, government authorities certify computer codes upon their validation (by themselves or by third parties).			X	"The user" should validate the code for the specific applications for which the DSA will be performed. The way it is validated is described in Chapter 5. Chapter 8 is related to "Documentation" and it is required that validation of the code "by the user" should be described. If this is done by another authority, it is the duty of the user to describe and to justify that. *(Note: maybe only two States certify the codes).*
Turkey 63	8.6 Line 3	"…safety analysis, with clear specification of conditions for applicability of the criteria (see	More informative to make connection with the relevant section of this	X			

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		Section 4 of this Safety Guide)."	Guide.				
Turkey 64	8.9 Line 5	"…margin, with no possibility of recriticality." Alternative: "…margin. Possibility of recriticality should be addressed."	Safe and stable end state needs to address the possibility of recriticality. Two suggestions are made here.	X	"… and the core is and will remain subcritical by a given margin."		
EC-JRC - 23 (Observer / uninvited)	8.9 Line 6	"… given margin~~.~~, and all the relevant safety variables, e.g. core and RCS temperature, RCS pressure, containment temperature, pressure, and flammable gas concentration, etc., has achieved a steady state such that no further significant changes are expected including cliff-edge effects.	Clarification		*Treated with Turkey 64*	X	It seems not necessary to enter into that level of detail in this Section/para.
Canada 80	8.14 Before para.	**Interface between safety and security regarding reporting of safety analysis** 8.14 Sensitive information …	Editorial. Should para. 8.14 be preceded by a heading "Interface between safety and security regarding reporting of safety analysis"?		*Paragraph 8.14 will be titled:* "Sensitive information of documentation"		
Canada 81	8.14	"… protected. This may include~~-includes~~ but is not limited to…"	Editorial	X			
Canada 82	8.17 Line 1	8.17. 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. …	Editorial. Delete "the".	X			
Canada 83	8.18 Line 1	8.18. The outcomes of the reassessment, including new deterministic safety analyses, if necessary, should be reflected in the	Editorial. Add comma after "reassessment".	X			

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		updated safety analysis report …					
Section 9							
Canada 84	9.2 Line 4	… safety requirements. As a minimum, it should be verified by the licensee ~~(but not necessarily limited to)~~ that the design will comply with the relevant regulatory requirements and acceptance criteria	Editorial. It is not necessary to have both "as a minimum" and "but not necessarily limited to" in the same sentence. Delete as indicated.	X			
Canada 85	9.6 Line 1	"…operating organization may choose to ~~should ensure~~…"	This is not consistent with current practices. Recommend changing "should" to "may choose to" in all instances in section 9, OR, add a new clause which indicates that independent verification by separate organizations is not mandatory.			X	GSR Part 4 clarifies that independent verification is not an option but an obligation.
IAEA/WES-36 (Unsolicited)	9.13 Bullet 3	• "Comparison with IAEA safety standards or other guidance documents;"	Editorial (same wording as in Para 9.18, 1st bullet).			X	Check - "Guidance documents" used in 9.13 (bullet 3) and 9.18 (bullet 1) has different meaning. In the first refers to IAEA safety requirements, safety guides and other IAEA guidance documents; in the second not necessarily IAEA guidance
IAEA/WES-	9.18	• "Assumptions and data used in	Usage of a consistent		*A term consistent with the Safety*		

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37 (Unsolicited)	Bullet 6	each analysis … are sufficient margins to prevent cliff-edge~~s~~ effects"	terminology throughout this Safety Guide is strongly recommended.		*Glossary will be used:* "… not lead to cliff edge~~s~~ effects. " *(See also IAEA/WES-21 to para. 6.7)*		
References							
Pakistan 4	10	*(Add new section):* **Quality Assurance of Deterministic Safety Analysis** Deterministic safety analysis shall be subject to a comprehensive QA program applied to all activities affecting the quality of the results. The QA program should identify the management system or quality assurance standards to be applied and should include documented procedures and instructions for the complete safety analysis process, including, but not limited to: a) collection and verification of NPP data b) verification of the computer input data c) validation of NPP and analytical models d) assessment of simulation results	A comprehensive QA program should be applied to all activities affecting the quality of the safety analysis results. This should be applied by personnel conducting independent review of entire analysis and results.			X	This aspect is covered by para. 3.8 of this Safety Guide (Management System).

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		e) documentation of analysis results					
IAEA/WES-38 (Unsolicited)	Page 58, Ref. [5]	"INTERNATIONAL ATOMIC ENERGY AGENCY, ~~A General Framework for~~ Prospective Radiological Environmental Impact Assessment ~~and Protection of the Public~~ for Facilities and Activities, Draft IAEA Safety Guide, DS427 Draft Version ~~7~~ 8.4, ~~August 2015~~ October 2016"	This is the title under which the Draft Safety Guide DS427 was endorsed for publication at the 40th CSS meeting in November 2016 (SPESS Step 12).	X			
IAEA/WES-39 (Unsolicited)	Page 59, Ref. [7] – [12]	Note: In the list of references, the publications [7] – [12] have to be renumbered to [12] – [17].	Wrong numbering of publications (the numeration of these references in the draft text itself is completely correct).	X	*(Numbering and description of all the References will be updated)*		
India 12	Page 59	References nos. from 12 onwards should be corrected to bring them in proper sequence	Editorial comment.	X	*(Numbering and description of all the References will be updated)*		
USA 1	References	The numbering of references is incorrect, with reference numbers [7] to [12] repeated. The reference for "NS-R-3" should be reference [12], and numbering should continue to [17].	Editorial	X	*(Numbering and description of all the References will be updated)*		
IAEA/WES-40 (Unsolicited)	Page 59, new Ref. [18]	In the list of references, please add a new Ref. [18]: "INTERNATIONAL ATOMIC ENERGY AGENCY, Severe Accident Management Programmes for Nuclear Power Plants, Draft IAEA Safety	For justification, see the related comments on Paras 2.18, 7.57 and A-25. DS483 is currently in SPESS Step 10.	X	*(Numbering and description of all the References will be updated)*		

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		Guide, DS483 (revision of IAEA Safety Standards Series No. NS-G-2.15 issued in 2009)**"**					
Annex							
Canada 86	A-1 Item (a)	(a) Design of nuclear power plants by the designer or verification of the design by the operating organization.	Editorial. Add "the" before "operating organization".	X			
ENISS - 13 (Observer / unsolicited)	Annex A1 (i)	(i) Demonstration of success criteria and development of accident sequences in Level 1 PSA and Level 2 PSA ~~the probabilistic safety analysis. Levels 1 and 2.~~	It is preferable to use the terms "Level 1 PSA" and "Level 2 PSA" (see SSG-3 and SSG-4)	X	*Bullet (i) will be modified as follows:* (i): Demonstration of success criteria and development of accident sequences in ~~the~~ Level 1 PSA (probabilistic safety assessment ~~analysis~~) and Level 2 PSA. *Title of subsection above A-27 will be modified accordingly*		
Turkey 65	A-1 Line 15	"… safety analysis (levels 1 and 2). ~~Levels 1 and 2.~~	Typo		*See ENISS 13*		
Canada 87	A-4 Line 4	"… relevant plant states and provides for adequate operating margins.	Editorial. Change "provide" to "provides".	X			
Turkey 66	A-6 Line 1	A-6 The operating organization usually performs or verifies the safety analysis to the extent necessary	Typo	X			
Canada 88	A-7 Line 3	"…It is therefore performed ~~in~~ with the same scope and following the same or even more stringent rules…"	Editorial. Change "in" to "with".	X			

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Turkey 67	A-13 Line 2	the findings of periodic safety reviews (when performed), changes in regulatory requirements, advances in	Better expression.	X			
Canada 89	A-18 Line 2	"…If there is a lack of detailed information on the plant ~~state~~ operating parameters, sensitivity studies, with the variation of certain parameters, may be performed.	"Plant state" refers to NO, AOO, DBA or DEC. What is intended here is "plant operating parameters".	X			
IAEA/WES 41 (Unsolicited)	Annex, A-22 Line 3	"…accident into a design extension condition~~s~~ with core melting."	Editorial.		*(See Canada 90 and Turkey 68)*		
Canada 90	A-22 Line 3	A-22 Deterministic safety analyses are also typically performed to assist the development of the strategy that an operator should follow if the emergency operating procedures fail to prevent progression of a design basis accident into ~~a~~ design extension conditions with core melting. …	Editorial. Delete "a" from before "design extension conditions".	X	*Same as Turkey 68*		
Turkey 68	A-22 Line 3	"…progression of a design basis accident into ~~a~~ design extension" conditions with core melting. The	Typo	X	*Same as Canada 90*		
Turkey 69	A-23 Line 1	A-23 The analyses are used to identify what challenges to the integrity **of** the barriers or alternative pathways for their by-pass can…"	Better expression.	X			
IAEA/WES-42 (Unsolicited)	Annex, A-25	"… The analysis supporting the development of severe accident management guidelines typically focus on mitigatory measures, which are strategies for managing severe accidents to mitigate the consequences	With a view to support structuring and to improve the readability of the entire sentence, please include consecutive numbering of the different strategies to		"... For water cooled reactors, such strategies may include: coolant injection into the degraded core; depressurization of		The strategies do not necessarily apply to all plants (e.g. existing); the term "may" seems more appropriate.

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
		of core melt. For water cooled reactors of different types that are in operation or being constructed, such strategies include: (a) coolant injection into the degraded core; (b) depressurization of the primary circuit; (c) activation ~~operation~~ of the containment spray~~s~~ system to prevent containment overpressurization and remove thermal energy from the containment atmosphere; and (d) use of the fan coolers, ~~hydrogen~~ passive autocatalytic recombiners for eliminating hydrogen from the containment atmosphere, and filtered containment venting [18]. ~~that are available in the reactors of different types that are in operation or being constructed.~~ …"	mitigate the consequences of a severe accident. The phrase at the end of the 3rd sentence should be shortened and moved to the beginning of this sentence because it fits better there. More detailed guidance on mitigatory measures during a severe accident is given in the Draft Safety Guide DS483 "Severe Accident Management Programmes for Nuclear Power Plants". Thus, a new reference [18] to DS483 should be added in Para A-25.		the primary circuit; activation ~~operation~~ of the containment spray~~s~~ system; ~~and use of the fan coolers~~, ex-vessel cooling of molten corium, ~~hydrogen~~ recombiners of combustible gasses and filtered containment venting [xx]. ~~that are available in the reactors of different types that are in operation or being constructed.~~ …"		
IAEA/WES-43 (Unsolicited)	Annex, A-26	"A-26 Transition from the emergency operating procedures to the severe accident management ~~procedures~~ guidelines, if they are separate, is to be carefully defined and analysed …"	In the Draft Safety Guide DS483 "Severe Accident Management Programmes for Nuclear Power Plants", the term 'severe accident management guidelines' (SAMG) is consistently used (see e.g. Paras 2.14 and 2.36 in DS483, Rev. 2 dated 31 July 2015); the same term is also found in Paras A-1 (h) and A-25 of DS491.	X			

COMMENTS BY REVIEWER				RESOLUTION			
Comment Nr	**Para / Line Nr.**	**Proposed new text**	**Reason**	**Accepted**	**Accepted, but modified as follows**	**Rejected**	**Reason for modification/rejection**
Turkey 70	A-26 Line 3	"…has guidance on the necessary actions and the monitoring of accident progression, ~~whatever~~ regardless of the"	Better expression.	X			
Turkey 71	A-28 Line 6	"…equipment failures and human errors, can prevent ~~prevents~~ nuclear fuel degradation. The deterministic analysis is"	Better expression.	X			

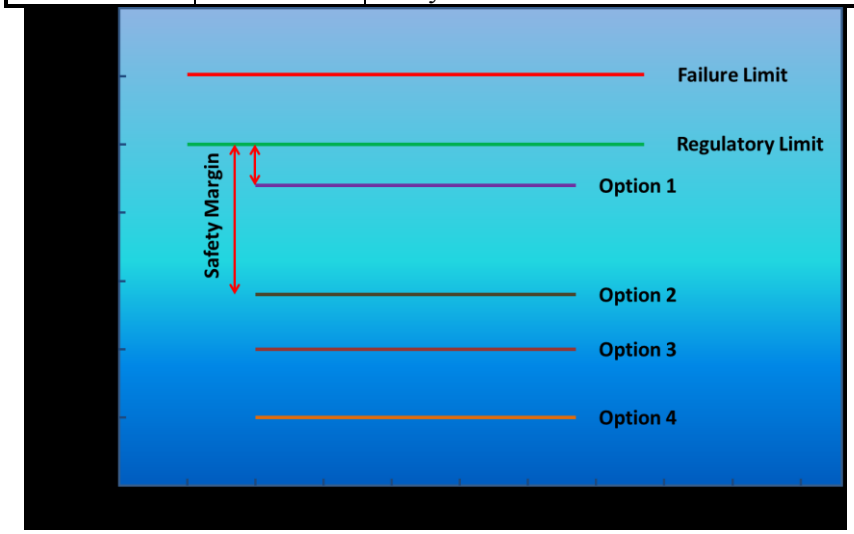

Figure 1 (Jordan)

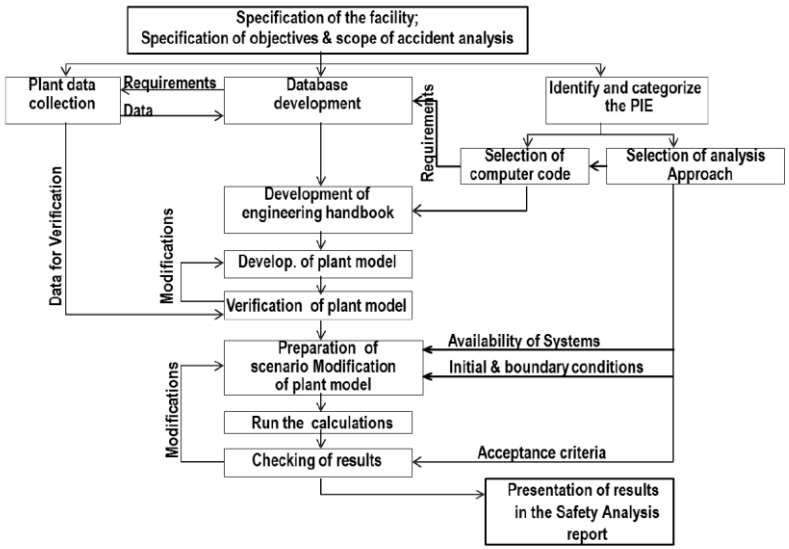

Figure 2

Figure 2 (Jordan)

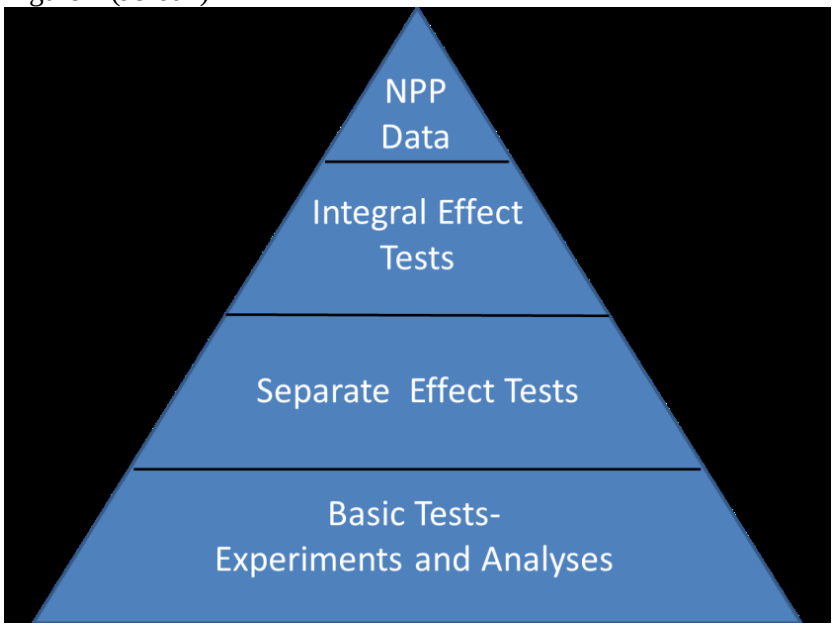

Figure 3 (Jordan)