Section 3
DS491 Step 7: Deterministic Safety Analysis for NPPs

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			X	*Table 2 of SSG2 (2009) seems outdated; it has been replaced by the data incorporated in § 3.26*

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

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

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

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

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

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

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

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d 2			results should be counted at first since it gives the deepest information of PIE's and their occurrences. Also, the engineering judgement in the establishment process of the design basis is to be mentioned as an important tool.		- Use of analytical methods …, failure modes and effects analysis (FMEA), engineering judgement and master logic diagrams		*at the beginning of a new design, it cannot be used as the major input for the list of PIEs*
Germany 7	3.19	3.19. The set of PIEs should be identified in a systematic way. This should include a structured approach to the identification of the PIEs such as: - Basis for the determination of the plant specific list of PIE should be the event spectrum determined by the vendor of the plant under examination; - Use of analytical methods such as hazard and operability analysis (HAZOP), failure modes and effects analysis (FMEA), and master logic diagrams; - Comparison with the list of PIEs developed for safety analysis of similar plants (ensuring that prior flaws or deficiencies are not propagated); - Analysis of operating experience data for similar plants; - Use of PSA Level 1 and Level 2 insights and results.	As starting point for the development of the plant specific list of PIE the event spectrum of the plant developed by the vendor of the plant which should be available should be used. After that, the list must be modified by using the following mentioned steps.				*Event spectrum determined by the vendor is typically used [should be] when it is available. Nevertheless, recommendations provided apply also to the vendor; it seems better not to identify vendor's list as an input.* *PSA use is out of the scope of this Safety Guide*
Czech 6	3.20	… accidents without careful analysis and assessment of the potential impact	dtto No5 comment			X	*See SSR2/1 Req. 20, §5.27*

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		on early ~~or~~ large releases.					
Observer ENISS-19	3.20	Ask for clarification	In some countries, certain limiting faults are excluded from the DBAs on the basis of specific justifications such as break preclusion approach. Req. 3.20 is not crystal clear. What is required? Is it required to demonstrate that the excluded events have a negligible contribution to the risk of large or early releases? The requirement should be written more clearly.		*It is indicated:* "… should not be excluded from this category of accidents without careful analysis…"		
Observer EC/JRC-35	3.20 Line 4	… accidents without careful **quantitative** assessment of the potential contribution to **overall risk figures of merit**.	Same reasons stated in previous comment 32		*Correction:* "…Secondary system pipe break…" *Last part made consistent with wording used in 3.7. See EC/JRC-33:* "… without careful analysis and quantitative assessment of its ~~the~~ potential of contribution to the overall risk, including to conditions arising that could lead to an early radioactive		

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					release or a large radioactive release”		
Switzerland 3	3.21	….normal operation should be considered as PIEs…..	As added	X			
Canada 10	3.21	3.21. Failures occurring in the supporting systems that impede the operation of systems necessary for normal operation should be also considered PIEs if such failures ~~eventually require the actuation of the reactor protection systems~~ directly lead to challenging safety functions and eventually to a threat to barriers against radioactive releases.	Most AOOs do not require actuation of the protection system, but they must nevertheless be analysed. Clauses 3.17 covers this already, but if it is necessary to repeat it, please use the same words.		…if such failures require protective actions		
Observer EC/JRC-36	3.21 Line3	the control and limitation systems.	According to the terminology used in 3.3, reactor protection system is included within the control and limitation systems. Unless distinction is wanted to be made here to limit PIEs related to supporting systems only to those leading to scram, same nomenclature should be used here.		*See Canada-10* …if such failures require protective actions		
Observer EC/JRC-37	*New*	Identification of PIEs applying to AOOs, DBAs and DECs should be carried out on a plant-operational-state basis.	Para 3.9 list of generic operational states should apply to the entire 'POSTULATED INITIATING EVENTS' section when talking about PIEs identification;				*See change in 3.22*

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			otherwise explicit mention in this respect should be made on PIEs identification dealing with AOOs, DBAs and DECs.				
Germany 8	3.23	3.23. All PIEs should be subdivided into representative groups of event sequences taking into account the expected frequency of occurrence and its effect on the nuclear power plant. This approach allows the selection of the same acceptance criteria and/or initial conditions in each group, applying the same assumptions/methodologies, and identification of the worst accident (bounding case) in each group.	It is not clear what is meant with "representative groups of event sequences". E. g. the German understanding is that for each level of defense an own set of acceptance criteria exists. The suitable set of acceptance criteria will be applied to each event grouped into the level of defense under examination. Does group mean level of defense? If yes, does the last sentence mean that only one bounding case should be analyzed for each level of defense? An adjustments of the expressions groups, categories, plant state (see table under 3.26) etc. used in the document should be adjusted.		*The content of §3.23 and §3.30 will be combined, resulting in the new §3.23 and §3.24 as follows.* *The wording of these two new paras also answers to other comments made about the same subject:* 3.23. ~~All~~ PIEs should be subdivided into representative groups of event sequences taking into account physical evolution of the PIEs. ~~the expected frequency of occurrence and its effect on the nuclear power plant.~~ These groups gather event sequences that lead to a similar threat to the safety functions and barriers and the need for similar mitigating systems to drive the plant to a safe state. Therefore they can be bound by a single representative sequence which is usually referred to when dealing with the		

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					group (and often identified by the associated PIE itself). Then these groups are also categorized according to their frequency of occurrence (see § 3.26). This approach allows the selection of the same acceptance criteria and~~/or~~ initial conditions ~~in each group, applying~~ and the application of the same assumptions and methodologies to all PIEs grouped under the same representative event sequence.~~, and identification of the worst accident (bounding case) in each group.~~ ~~3.30. Groups of PIE should be further subdivided according the mechanisms affecting the performance of the safety functions and integrity of the physical barriers. Special groups of sequences can be thus formed~~ 3.24 Representative event sequences can also be grouped by type of sequences with focus on reduced core cooling and RCS pressurization,		

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					containment pressurization, radiological consequences, or pressurized thermal shocks. For instance the PIEs "stop of a MFW pump", "stop of all MFW pumps", "isolable break on MFW system" are all typically grouped under a single representative event sequence which is "Loss of Main Feed Water" which belongs to the "Decrease in reactor heat removal" type of sequence.		
Czech 7	3.23	All PIEs should be subdivided into representative groups of event sequences taking into account the expected frequency of occurrence and its effect on the safety of the nuclear power plant.	Text clarification.		*See resolution to Germany-8*		
Observer EC/JRC-38	3.23/2	… and its effect on the nuclear power plant**, i.e. similar mitigating systems needed to drive the plant to a safe state**.	'effect on the nuclear power plant' should be clarified.		*See resolution to Germany-8*		
Germany 9	3.24	3.24. The postulated initiating events associated with anticipated operational occurrences and DBAs should reflect specifics of the design, but typically should belong to the following types of transients: - Increase or decrease of the heat removal from the RCS; - Increase or decrease of the RCS	The list of event categories should be expanded as shown.		*Now §3.25* - Leaks ~~inside and~~ outside containment;		- Increase or decrease of the RCS pressure; *(Already covered by increase/decrease in heat removal and increase/decrease of RCS inventory)*

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		flow rate; Increase or decrease of the RCS pressure; Anomalies in reactivity and power distribution in the reactor core or in the fresh or spent fuel storage; Increase or decrease of the reactor coolant inventory; Leaks in RCS without/with potential containment by-pass; Leaks inside and outside containment; Loss of offsite power; Reduction or loss of cooling of the fuel in the spent fuel storage pool; Release of radioactive material from a subsystem or component (typically from treatment or storage systems for radioactive waste).					Leaks in RCS without/with potential containment by-pass; *(Without is a LOCA; previous bullet)* Loss of offsite power; *(Covered by other bullets (decrease of RCS flow, decrease of the heat removal)*
Canada 46	3.24	Suggest an additional bullet, Loss of cooling to fuel during on-power refuelling for PHWR	For PHWR, loss of cooling during on-power refuelling should be considered.	X	*Now §3.25*		
Observer EC/JRC-39	3.24,3.27,3.29/All	Identification of PIEs can be made by attending to events related to challenging different critical safety functions. Within each category of events, PIEs are identified according to plant-specific features. Typical examples of category of events challenging safety functions are the followings:	First, examples shown in 3.24 are classified as "*types of transients*". However, 3.24 and entire section 3 talks about PIEs, i.e. initiating events, so the guide should keep referring to events rather than transient, where the latter could also embrace, as indicated in para 3.23 assumptions and		3.24. The postulated initiating events associated with anticipated operational occurrences and DBAs should reflect the specifics of the design. ~~, but typically should belong to the following types of transients:~~ Some typical PIEs and		

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

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					The actual list will depend on the type of reactor and actual design:		
Observer ENISS-20	§3.24/3rd bullet	Anomalies in reactivity and power distribution in the reactor core or in the fresh or spent fuel storage unless these are practically eliminated as presented in paragraphs 7.68 to 7.72 of this Safety Guide;	For the Fuel Building, the safety demonstration associated to reactivity anomalies is based on criticality safety principles with a dedicated referential. As such, they follow a different approach from deterministic studies and should be excluded from this guide.			X	*Practical elimination can be claimed but PIE has to be considered*
Observer EC/JRC-40	3.24	*Remove entire para 3.24*	Para 3.24 is nearly redundant with para 3.27 and 3.29. It does not say anything not accounted for in the other two referred paras.			X	
Czech 8	3.25 Line 2	Special attention should be paid to accidents in which the release of radioactive material could bypass the containment because of potentially large consequences even in the case of relatively small releases of radioactive substances from the core.	Specification of what releases are in mind.			X	*Seems unnecessary*
Canada 11	3.25, last sentence	Moreover, ~~such~~ large bypass accidents do not allow much time for taking action to protect the public in the vicinity of the plant.	Small bypass accidents allow plenty of time to protect the public.	X	Clarification		
Switzerland 4	3.26	Table: DBA Limiting Faults DBC-4, PC-4	It should clearly stated that for existing power plants this frequency range was			X	*This consideration applies to other aspects of the SG.*

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

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

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

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Pakistan 3	3.27 Bullets 4 and 5 Page 13	Typical examples of PIEs for "Reactivity and power distribution anomalies in the fresh or spent fuel storage" are missing.	Examples of PIEs for "Reactivity and power distribution anomalies in the reactor core" are addressed. In a similar way, PIEs for fresh fuel or spent fuel storage may also be mentioned for completeness and invigorating better understanding.		*Now §3.28* *Bullet 4:* - Reactivity and power distribution anomalies in the reactor core: inadvertent *New bullet 5:* - Reactivity anomalies in the fresh or spent fuel storage: dilution in spent fuel pool *Bullet 7 (now 9):* - ~~Failures of systems ensuring normal operation of fuel pools:~~ Reduction or loss of fuel cooling in the SFP: loss of off-site power… *Bullet 8 (now 10):* - Release of radioactive material ~~from~~ due to leak in RCS with potential containment bypass or from a subsystem or component: minor…		
Japan 5	3.27, 1st bullet	—Increase in reactor heat removal: inadvertent opening of steam relief valves; ~~secondary~~ pressure control	Generalization to include BWR plant.	X			

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		malfunctions leading to an increase in steam flow rate.					
Japan 6	3.27, 3rd bullet	—Decrease in RCS flow rate: trip of ~~one main coolant pump~~ one or more coolant pump(s); inadvertent isolation of one main coolant system loop (if applicable).	Generalization to include BWR plant.	X	*Bullet 3:* —Decrease in RCS flow rate: trip of one or more coolant pumps; inadvertent isolation…(if applicable); start of a main coolant pump		
Canada 47	3.27	Suggest additional example of PIE for PHWR Loss of moderator circulation or decrease or loss of moderator heat sink for a PHWR	For PHWR, moderator system malfunction is an important AOO.	X	*New bullet 6*		
Observer ENISS-22	3.28 - line 2	The subset of PIEs leading to DBAs should be identified. ~~All PIEs identified as initiators of anticipated operational occurrences should also be considered as potential initiators for DBAs.~~ Although … specific reactor.	PIEs identified as initiators of AOOs cannot be DBAs. AOO PIEs consist in frequent events associated to the failure of normal operating functions (as shown with examples given in 3.27) whereas DBA PIEs consist in less frequent events associated with pipe breaks (as shown with examples given in 3.29). As written in 7.33, *"an anticipated operational occurrence by itself should not generate a DBA"*.		"The subset of PIEs leading to DBAs should be identified. All PIEs identified as initiators of anticipated operational occurrences should also be analyzed using DBA rules (see SSR2/1 § 5.75(e)). Although … specific reactor."		

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

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

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

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

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		material;	or storing.				
Observer ENISS-26	§3.35	In addition, there are … include: (a) A reduction in or loss of cooling of the fuel in the spent fuel pool (if leading to boiling); (b) Reactivity anomalies in the fresh or spent fuel unless practically eliminated as presented in paragraphs 7.68 to 7.72 of this Safety Guide;	(a) In most cases, a partial loss of cooling in the spent fuel pool does not lead to boiling and as such, does not lead to any radiological release. (b) Is the word "storage" missing? In any case, a reactivity anomaly in the fresh or spent fuel storages leading to the release of radioactive material would correspond to a criticality accident, that has to be excluded, and for which the source term is difficult to assess.			X	*See ENISS-20 to 3.24.* *Practical elimination can be claimed but PIE has to be considered*
Germany 14	3.36	3.36. The frequency associated to a type of anticipated operational occurrences or DBA should combine the frequencies of all PIEs that have been grouped together.	The link to frequencies is not clear. Furthermore, the relevance of determination of frequencies of PIEs in the frame of deterministic event analyses is not clear. An adjustment of the expressions is needed (see comment 8) because it is unclear what "grouped together" means.	X	3.36. The frequency associated with a bounding event sequence belonging to ~~to a type of~~ AOO or DBA should use the bounding frequency established for the ~~combine the frequencies of all~~ PIEs that have been grouped together. *Note: 3.26 could be merged also with the new 3.23-3.24 (which*		

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

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

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

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

Comment No.	Para/Line No.	Proposed new text	Reason	Accep ted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
					mitigating core melting accidents, according to the plant safety objectives		
Observer EC/JRC-46	3.45	*Same comment than 45* (just by replacing Level 1 by Level 2).	Same rationale than 45 applies here but applied to DEC with core melting.			X	*DEC-B is deterministic and corresponding conditions are postulated regardless the estimated frequency. The major physical phenomenon have to be addressed*
Germany 16	3.46	3.46. Deterministic safety analysis should consider that the features to prevent core melting fail or are insufficient and an accident sequence will further evolve into a severe accident. Some representative sequences should be selected by adding additional failures or incorrect operator responses to the DBA or design extension conditions sequences, ~~and to~~ by using the dominant accident sequences identified in the Level 2 PSA and by selecting scenarios of Level 2 PSA with large releases independently from their frequencies.	For selection of possible sequences also scenarios of Level 2 PSA with large releases should be considered independently from a very low frequency.			X	*DEC-B is deterministic and corresponding conditions are postulated regardless the estimated frequency. The major physical phenomenon have to be addressed* *Additionally, PSA is not available at the beginning of the design, when severe accident conditions have to be defined. Selection of scenario independently of frequency is mentioned in § 3.49*

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

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

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

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

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

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		failure of shared equipment.			ultimate heat sink and the failure of shared equipment should be taken into account."		
Switzerland 5	3.53	- such hazard can be screened out due to its negligible contribution to risk, or	A definition of "negligible contribution to risk" should be added or referenced.		"3.53 The analysis of hazards which is performed by using probabilistic methods or appropriate engineering methods [a Reference will be added] should demonstrate…"		*Definition of "negligible contribution to risk" is out of the scope of this SG but has to be assessed in [hazard] dedicated guides*
Czech 12	3.55	Event sequences that lead to early or large radioactive releases[5] are required to be practically eliminated	Use this (5) below the line explanation in para 2.1 where wording early or large is used for the first time, if my comment against using his wording "early or large" will not be accepted..		*See other resolutions, e.g. 2.1 (CZ-2 and other) and 2.18.* "…minimized. Conditions arising that could lead to an early radioactive release or a large radioactive release"…		
Observer EC/JRC-50	*After* 3.55/New	According to SSR-2/1, Rev. 1, two types of source term release scenarios should be 'practically eliminated': large release and early release category. Since severe accident consequences on source term magnitude, composition and timing to determine whether a particular scenario should be classified under one of the two abovementioned categories is a very complex issue,	Para 3.56 should be deeply improved: First, classification attending to 'events' and 'severe accident phenomena' does not fit well with identifying conditions leading to large or early release. Second, it is not mentioned how plant-specific this issue			X	*The clarification seems not necessary and out of the scope of this SG. On the other hand it is severe accident oriented; some DBA are also excluded because of the practical elimination*

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

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

Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
		may lead to severe source term releases in the long term as a consequence of losing the last defence-in-depth barrier.					
Germany 17	3.56, Line 1, page 19	3.56. The event sequences requiring specific demonstration of their "practical elimination" should be classified as follows, ~~if need be with a design specific adaption~~:	Is the colored part of the sentence necessary? Deletion improves readability.	X			
Japan 12	3.56 (group 2)	2) Severe accident phenomena which could lead to early containment failure: a. Direct containment heating b. Large steam explosion c. ~~Hydrogen detonation~~ Large hydrogen explosion	It is not ensured solely detonation will lead to containment failure.		"c. Explosion of combustible gases, including hydrogen and carbon monoxide"		
Japan 13	3.56 (group 3)	3) Severe accident phenomena which could lead to late containment failure: a. Molten core concrete interaction (MCCI) b. Loss of containment heat removal c. Large hydrogen explosion	Hydrogen explosion is not limited in early phase.	Partially	"c. Explosion of combustible gases, including hydrogen and carbon monoxide"		
Germany 18	3.56, Grop 3 Page 20	3) Severe accident phenomena which could lead to late containment failure: ~~a. Molten core concrete interaction (MCCI)~~ b. Loss of containment heat removal	Current experiences with severe accident analyses for different reactor types have shown that late failure of containment by MCCI cannot be practically eliminated, especially for older plant designs. May be for next generation plants		*See France-11*		*It should be practically eliminated in new design, otherwise radiological consequences of SA cannot be limited*

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			like EPR that might be possible. Checking if that can be listed here as a scenario which can be practically eliminated.				
France-11	3.56 2)	2) Severe accident ~~phenomena which~~ events that could lead to early containment failure: a. Highly energetic direct containment heating b. Large steam explosion c. Hydrogen detonation or deflagration with impacts exceeding the containment capacity 3) Severe accident conditions ~~phenomena~~ which could lead to late containment failure FNXX: a. Basemat penetration or containment bypass during molten core concrete interaction (MCCI) b. Long term loss of containment heat removal leading to an uncontrolled failure of the containment 4) Severe accident with containment bypass 5) Significant fuel degradation in a storage pool and uncontrolled release FNXX – These conditions should be analysed during the identification of situations to practically eliminate. Nevertheless, it should be generally practicable to mitigate them.	It should be better to consider in the safety analysis severe accidents which could lead to late containment failure and to mitigate them according to DiD because for most of them,in particular for new reactors, mitigation is possible The text above is not related to phenomena		*See Japan 12 and 13* 2) Severe accident sequences that ~~phenomena which~~ could lead to early containment failure: a. Highly energetic ~~D~~direct containment heating b. Large steam explosion c. ~~H2 detonation~~ Explosion of combustible gases, including hydrogen and carbon monoxide 3) Severe accident sequences that ~~phenomena which~~ could lead to late containment failure: a. Basemat penetration or containment bypass during ~~M~~molten core concrete interaction (MCCI) b. Long term of ~~L~~loss		- *"events" in 2 and "conditions" in 3 harmonized (events is used in the SG).* - *It seems better to use "explosion" only in this SG and not also deflagration and detonation.* - *2 (c) "impacts exceeding the containment capacity" is included in the title. Similar for 3 (b)* - *Foot note seem out of the scope of 3.56*

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

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			dedicated SAMG aimed at such scenario? Paras 7.68-7.72 describes pertinent suggestions to conduct deterministic analysis for ' practically eliminated' scenarios. But these rules should not ever been referred as 'severe accident management guidance' for obvious reasons.				