MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Hungary	8	General	Recommendations for integrated PSA models	There should be a set of recommendations in a specific chapter or subchapter for integrated PSA models that does not use the PDS approach.			X	Recommendations for the integrated approach for Level 2 PSA model related to PDS are presented in Section 5.
Israel	1	General ("unusual") remark regarding DS 528:	Usually, our comments include suggestions for clarifications, or for adding details for completeness. In the present case, after thorough reviewing of draft safety guide 528, we would like to point out that we find this guide very well and clearly drafted. It is an important and useful safety document. For not leaving the compliment comment "lonely" on this Form for Comments, please find below a few suggestions and also some editorial nature remarks:	**Complement**	X			Thank you for the compliment.
Senegal	1	General	No comments		X			
Germany	1	1,2	Several IAEA Safety Requirements publications establish general and specific requirements on risk assessment for nuclear power plants. Paragraph 4.13 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [2]) states:…	Clarification. Same for paras 1.4, 1.14, 1.21. Please check whole text.	X			
ENISS	1	1,5	~~Thus, a full scope probabilistic safety assessment (PSA) will contribute to assess and verify the safety of nuclear power plants in relation to potential internal initiating events and internal and external hazards as well as their combinations.~~ Thus, probabilistic safety assessment (PSA) is considered to be an important tool for analysis to ensure the safety of a nuclear power plant in relation to potential initiating events that might be caused by random component failure or human error, as well as by internal and/or external hazards.	A more general text is proposed, without any mention to a "full scope PSA" which is not defined so far (defined later in the para 1.18) and not required by the § 5.76 of SSR-2/1. The text proposed is from DS523 (SSG-3 pre-print version).		X…Thus, a probabilistic safety assessment (PSA) will contribute to assess and verify that a balanced design of the nuclear power plant has been achieved in relation to the overall risk from potential internal initiating events and internal and external hazards, and to prevent cliff-edge effects.		Agree to modify and to comly with para 5.76 of SSR-2/1 (Rev.1), therefore balanced design, risk and cliff-edge needs to be added.
Japan	1	1,5	Thus, a ~~full scope~~ probabilistic safety assessment (PSA) will contribute to assess and verify the safety of nuclear power plants in relation to potential internal initiating events and internal and external hazards as well as their combinations.	PSA other than full scope PSA may contribute.		X… Thus, a probabilistic safety assessment (PSA) will contribute to assess and verify that a balanced design of the nuclear power plant has been achieved in relation to the overall risk from potential internal initiating events and internal and external hazards, and to prevent cliff-edge effects.		Agree to modify and to comly with para 5.76 of SSR-2/1 (Rev.1), therefore balanced design, risk and cliff-edge needs to be added.
Germany	2	1,6	… Further guidance ~~information~~ is provided in IAEA Safety Standards Series No. SSG-3, Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants [4].	Please check if wording "guidance" or "recommendations" is more suitable here	X			
Israel	2	1,6	**In paragraph 1.6 (1), Level 1 PSA is mentioned regarding design and operation of the plant being analysed in order to identify the sequences of events that can lead to core/and or fuel damage and the corresponding core and or fuel damage frequencies are estimated. The core and/or fuel damage are also mentioned regarding Level 2 PSA in paragraph 1.6(2). However, in paras. 1.8(a) and 1.8(b), mentioning Level 2 PSA, only fuel damage is referred to, without mentioning core damage. We suggest to consider this point (considering that, for example in paragraph 1.9(a) core and or fuel damage are mentioned together, again.**	Clarity	X			
Germany	3	1,7	PSAs are also classified according to the range of initiating events (internal and/or external to the plant) and plant operatin~~g~~onal modes that are to be considered.	Please change to "operational modes", to be in line with IAEA Glossary.		X...and modes of operation of the plant...		To be in compliance with SSG-3 (Rev. 1)
Canada	1	1,8	"(a) Level 1 PSA provides information on the accident sequences that lead to core and/or fuel damage…" "(b) The interface between Level 1 PSA and Level 2 PSA is where the accident sequences leading to core and/or fuel damage…"	To be consistent with Par 1.6 and across the SSG.	X			
ENISS	2	1,8	b) The interface between Level 1 PSA and Level 2 PSA is where the accident sequences leading to fuel damage are grouped into plant damage states (PDSs) based on similarities in the plant conditions that determine the further accident progression. ~~If the status of SSCs dedicated to ensuring the confinement function was not addressed in the Level 1 PSA, it needs to be considered by means of so-called 'bridge trees' of the interface between Level 1 PSA and Level 2 PSA or by extended Level 1 event trees, as the first step of the Level 2 PSA~~	This "need" of bridge-trees primarily concerns L2 PSA developed using a separated approach (see Para. 5.6) In order to have a general text (applicable to both approaches: integrated or separated) and consistent with para 2.9, a removal of this text is proposed.		X Some extended event trees can complete the information provided by Level 1 PSA….		To be in complieance with the Fig1.
Finland	1	1,8	1.8 Fig. 1 Add "Fuel Damage Frequency" in the box concerning results of level 1 PSA.	The level 1 PSA risk metric associated with fuel damages outside of the reactor core is usually called "Fuel Damage Frequency".	X			
Japan	2	1,8	Level 2 PSA is a structured process. Although there may be differences in the approaches for performing a Level 2 PSA, e.g., an integrated approach, see para 2.6, the general main steps are shown in FIG. 1 and are as follows:	To be consistent with or to clarify the relation to the statement of para 2.6.		X...Level 2 PSA (see para 2.6), the		There is no need to mention 1 approach over the other, but reference to para 2.6 is added.

Japan	3	1,8	… (b) The interface between Level 1 PSA and Level 2 PSA is where the accident sequences leading to fuel damage are grouped into plant damage states (PDSs) based on similarities in the plant conditions that determine the further accident progression. If the status of SSCs dedicated to ensuring the confinement function was not addressed in the Level 1 PSA, it needs to be considered by means of extended Level 1 event trees or by so-called ‘bridge trees’ of the interface between Level 1 PSA and Level 2 PSA or by extended Level 1 event trees, as the first step of the Level 2 PSA. ...	Since “extended Level 1 event tree” was identified as the linkage between Level 1 PSA and Level 2 PSA in FIG. 1, the term should be explained the first.			X	Text deleted based on comments from Pakistan and Canada,
Japan	5	1,8	“LEVEL 1-2 INTERFACE” in FIG.1 “EXTENDED LEVEL 1 EVENT TREE”	As described at paras. 1.8 (b), 8.13, etc.			X	The text refers to the extension of Level 1 PSA to Level 2 PSA, but it is part of Level 2 PSA.
Pakistan	1	1,8	Level 1 PSA provides information on the accident sequences that lead to **core and/or**fuel damage…	As per figure 1 of draft report, the end states are defined as core/fuel damage, therefore the suggested text may be added.	X			
Pakistan	2	1,8	…accident sequences leading tocore and/orfuel damage are grouped into plant damage states	As per figure 1 of draft report, the end states are defined as core/fuel damage, therefore the suggested text may be added.	X			
Mexico	1	1,9	Some methodologies ~~us~~ use a multi-step process	Improve wording	X			
Germany	4	1,10	Further grouping or regrouping of the release categories into a condensed set2 that would be taken forward into the Level 3 PSA may be needed. The interface between Level 2 PSA and Level 3 PSA is not addressed in detail in this publication ~~document~~ although it is touched upon in section 15 on the use and applications of Level 2 PSA.	Editorial	X			
Germany	5	1,12	This Safety Guide was prepared on the basis of a systematic review of relevant IAEA publications, including Refs [1][6] and an International Nuclear Safety Advisory Group (INSAG) report [7].	Editorial	X			
Bulgaria	1	1,16	~~For example,~~ for Molten Salt Reactors with liquid fuel, the concept of core melt is not meaningful. On the other hand, for Molten Salt Reactors with liquid fuel, the concept of core melt is not meaningful.	The sentence explains about the possible inapplicability of the guidance. Therefore, the connection of the last sentence to the previous one should express opposition and not example of the previous statement.			X	The sentence gives an example of phenomelogy related to a particular reactor technology which needs to be reconsidered when applying the recommendations in this safety guide. It is not an opposition but a follow up of previous sentence.
Japan	4	1,16	Most of the phenomenology described as examples in this Safety Guide is directly applicable to current water cooled reactors, such as pressurized water reactors or boiling water reactors but the phenomenology should be investigated and identified for ~~may also apply or adapt to~~ a particular reactor technology. For example, for Molten Salt Reactors with liquid fuel, the concept of core melt is not meaningful.	Level 2 PSA methodology described in this safety guide is applicable to other reactor technologies than water cooled reactors. But the phenomenology depends on the reactor technologies.	X			
Bulgaria	2	1,18	if the scope of the Level 1 PSA is limited to CDF/FDF assessment (see paras 2.9~~8~~-2.10~~9~~), additional analysis ~~to that described in this Safety Guide~~ may need to be carried out.	First, it should be clarified what kind of limitation of L1 PSA is envisaged here, then 2.9 and 2.10 seem to be more appropriate paras. and third, if by “additional analysis” are meant those related to containment and containment system, then they are part of this guide.		X Sentence deleted.		The sentence does not provide clear guidance on the applicable recommendations in the safety guide than those already mentioned in previous sentence "If the objectives and the scope of the Level 2 PSA are limited, only the relevant recommendations provided in this Safety Guide apply."
Germany	6	1,19	… Such an aim is not detailed in this Safety Guide, which focuses on releases of radioactive material resulting from severe accidents in the reactor and the spent fuel pool (SFP).	If the abbreviation SFP is to be used in this Safety Guide, please insert it here, as this is the first appearance of the full term in the text.	X			
Bulgaria	3	1,21	Recommendations relating to the performance, project management, documentation and peer review of a PSA and implementation of a management system in accordance with IAEA Safety Standards No. GSR Part 2, Leadership and Management for Safety [5] are provided in SSG-3 (Rev. 1) [4] and are therefore not addressed in detail here	The clarification is needed, since actually a lot of aspects are discussed here, in this safety guide, considering performance, project management, documentation and peer review.	X			
Saudi Arabia	1	1,22	[…] an overview of human reliability *analysis* in Level 2 PSA Human& equipment reliability assessment (section 8)	Please consider consistently using in all the text either “human reliability analysis” or “human reliability assessment”, noting that “human reliability analysis” is used in the previous IAEA documents.	X			
China	1	2	Section 2	It is recommended to add methods or practices to determine the probabilistic safety goal or risk criteria for multi-unit site in Section 2.			X	Recommendations related to defining risk metrics for Level 2 PSA are provided in section 2 for single NPP and in para 14.8 for multi-unit NPP. The intention of the safety guide is not to provide methdos or practices on this particular point. References to IAEA relevant publications on methods and practices for multi-unit Level 2 PSA are added in para 14.1.
China	2	2	Section 2 &3 The title “Scope of the Level 2 PSA Project” is suggested to be changed as “Determination of the Scope of the Level 2 PSA Project”	To be separated with the similar content in chapter 2 “Scope of Level 2 PSA”, which is more concentrated in the technical aspect, the content here in chapter 3 focus on the project management aspect.	X			
Israel	6	2	Page 7: Pages numbering starts on page 1, however it starts again as page 1 on the seventh page of the draft document.		X			
Japan	6	2,2	The main objective of Level 2 PSA is to determine if sufficient safety provisions have been made to manage a severe accident and to mitigate the effects of such an accident to ensure that sufficient protection of the population and the environment has been achieved. ~~and, for~~ For new reactor designs, Level 2 PSA, in combination with Level 1 PSA, contributes to demonstrating the ‘practical elimination’ of plant event sequences that could lead to an early radioactive release or a large radioactive release (see also IAEA Safety Standards Series No. SSG-88, Assessment of the Safety Approach for Design Extension Conditions and Application of the Practical Elimination Concept in the Design of Nuclear Power Plants [9].	Demonstration of the practical elimination of a specific accident sequence should rely on both Level 1 and Level 2 PSA.	X			

Japan	7	2,2	(a) Systems provided specifically to mitigate the effects of the severe accident, such as in-vessel or ex-vessel molten core retention features, hydrogen mixing devices or hydrogen recombiners, or filtered containment venting systems for water cooled reactors;	To clarify that these examples are for water cooled reactors			X	Not all examples are applicable only for water cooled reactors, e.g. in-vessel retention is a strategy considered for Liquid Metal Cooled Fast Reactors.
Russian Federation	1	2,3	(n) To provide base list of representative severe accidents for deterministic analysis.	Ensuring the completeness of the objectives of the Level 2 PSA. As the objectives of level 2 PSA, it is proposed to add in paragraph 2.3: "To provide base list of representative severe accidents for deterministic analysis". It does not follow from paragraph 2.3 (a) the need to form a list of representative scenarios of severe accidents. It is obvious that Level 2 PSA is the very tool that, among other things, determines the list of severe accidents to be taken into account in the NPP design (in detenninistic analysis).		X… n) To inform the choice of representative severe accidents for deterministic analysis.		Text modified for better reading.
Germany	7	2,5	… ~~While~~ Furthermore, Requirement 14 of GSR Part 4 (Rev. 1) [2] states that "The performance of a facility or activity in all operational states and, as necessary, in the post-operational phase shall be assessed in the safety analysis.	Editorial	X			
Saudi Arabia	2	2,5	"A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out at a particular stage for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity", while Requirement 14 of GSR Part 4 (Rev. 1) [2] states that "The performance of a facility or activity in all operational states and, as necessary, in the post-operational phase shall be assessed in the safety analysis."	The use of "while" supposes to have the two sentences in only one sentence.			X	There are two sentences joined by "while" because there are two requirements.
Bulgaria	4	2,6	~~In the separated approach, the Level 2 PSA is performed after the Level 1 PSA is complete, when some additional system analyses may be necessary.~~	Delete the statement. In both approaches, L2 PSA can be performed after L1 PSA. It is a matter of organization and not related to the adopted approach.	X			
Bulgaria	5	2,6	~~If the Level 2 PSA is performed following an integrated approach, the requirements of the Level 2 PSA should be fed into the Level 1 PSA; in this way, all plant related features that are important to the analysis of the response of dedicated SSCs ensuring the confinement function and the analysis of the source terms will be considered wherever possible in the Level 1 PSA~~	Delete the statement. The text is confusing and is better to be deleted. If changes in L1 PSA model are introduced in order to meet requirements of L2 PSA, i.e. adding new systems and etc., this should be done in a way that will not impact the L1 original model. The calculated CDF and FDF should be guaranteed. Otherwise, this will cause an overestimation of the CDF, FDF (e.g., in case that in the main L1 PSA ETs are included excessively to CDF/FDF systems that are needed for L2 PSA). These plant features should be addressed in L2 PSA (as part of bridge trees or APET). One more option is to have extended L1 PSA where in addition to CDF and FDF, containment status and systems are modelled.	X			
ENISS	3	2,6	Proposal to move this para 2.6 after para 5.1 (or eventually after para. 1.11).	This para. 2.6 does not deal with the scope of Level 2 PSA. Moving this para. seems preferable. Warning: many other para refers to this current para 2.6.		X Title changed to "Scope and approaches for Level 2 PSA"		To cover the introduction of the approaches.
ENISS	4	2,6	If the Level 2 PSA is performed following an integrated approach, the requirements of the Level 2 PSA ~~should~~ **may** be fed into the Level 1 PSA; in this way, all plant related features that are important to the analysis of the response of dedicated SSCs ensuring the confinement function and the analysis of the source terms will be considered wherever possible in the Level 1 PSA.	In an integrated approach, information about SSCs ensuring the confinement function remain directly accessible for the Level 2 model (see para 5.7). It is a possibility to feed such requirements of the Level 2 PSA into the Level 1 PSA but it shouldn't be a recommendation.		X Para deleted.		Considered together with comment 5 Bulgaria
Germany	8	2,6	…. The second is a separated approach, where the Level 1 and Level 2 PSA models are not developed, linked or quantified in a single software tool such that additional steps to transfer data ~~/~~, information ~~/~~ and results from Level 1 to Level 2 would be required. Reference [21] from the ASAMPSA2 project provides information on the advantages and disadvantages of each approach ~~[21]~~.	Clarification	X			
France	1	2,8	Commonly, the Level 2 PSA is developed as a base model for a comprehensive list of PDSs related to internal events. This base model should be used for the extension to ~~relevant operating modes and to~~ internal and external hazards (see Ref. [10]).	Internal event L2 PSA should cover all reactor operating modes	X			
Hungary	1	2,8	Commonly, the Level 2 PSA is developed as a base model ~~for a comprehensive list of PDSs~~ related to internal events. This base model should be used for the extension to relevant operating modes and to internal and external hazards (see Ref. [10]).	As stated in previous recommendations in newer PSAs the level 1 and level 2 PSA models are integrated and technically skip the PDS categorization process, hence the recommendation would be useless to many new PSAs without removing the highlighted text. By simply stating that the level 2 PSA models are first developed for internal initiating events the meaning and the purpose of the recommendation remains, while the exclusive part is removed.	X			
ENISS	5	2,9	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 2.9… containment and its associated systems …		In compliance with SSG-53.

Finland	2	2,9	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "associated systems"		The term "safety systems" in this context refers to containment and associated systems in accordance with the SSG-53.
Saudi Arabia	3	2,9	[…..] containment ~~safety~~ systems	Not all the concerned systems are safety systems.		X … containment and its associated systems		Same text as in SSG-53 "Design of the Reactor Containment and Associated Systems for Nuclear Power Plants"
Bulgaria	6	2,11	...the Level 2 PSA should consider the simultaneous consequences of severe accident phenomena induced by the reactor core and the spent fuel pool for this containment and the source term calculations. Recommendations on Level 2 PSA for the spent fuel pool are provided in section 13 of this Safety Guide.	The guidance in section 13 should be expanded to cover all issues related to simultaneous consequences. Note that simultaneous accident progression is considered only from the PDS point of view (see 13.7). This is found to be insufficient. See comments below.		X...the Level 2 PSA should consider combined consequences of severe accident phenomena induced by the reactor core and the spent fuel pool …		The severe accident consequences of reactor core and spent fuel pool could have combined consequences for the containment despite accident scenarios in the reactor core have different timing than for the spent fuel pool. The modification in the text of para 2.11 and 13.7 are presented to highlight those possible consequences.
Hungary	2	2,12	If the scope of the PSA includes sources of radioactivity other than the reactor and the spent fuel pool (e.g. refueling pool, transport casks, liquid radioactive waste or dry long-term storage of spent fuel) located outside of the containment (e.g. reactor containment building), then the potential risk of release from those sources should be considered. As stated in para 1.19, releases from other sources of radioactivity from the plant, such as irradiated fuel and stored radioactive waste, are not detailed in this safety guide.	I think it should be highlighted, that the recommendation refers to radioactive sources beyond the RPV and the SFP. I also believe that that two major alternative source of radioactive releases (refueling pool and transport casks) should be included in this list.	X			
Germany	9	2,15	If several reactor units (~~e.g.~~ i.e. power and/or research reactors) or reactor units and other radionuclide sources are collocated at the site, the scope of the Level 2 PSA might include the impact of severe accidents for the accident management of more than one unit on the site and the corresponding aggregation of risk for these units and/or sources on the site.	1) Please check if abbreviation i.e. is more suitable here. 2) The multi-unit aspect is also valid to be a multi-source issue and therefore should be addressed.			X	In accordance with para 1.19 the scope of this safety guide does not cover other radioactive sources.
France	2	2,18	Therefore, in defining the Level 2 PSA risk metrics and especially the terms “large”, “early”, “release”, “exceedance” and “frequency” should be considered~~, as follows~~:	Wording and consistency with the the following text (a) to (f) r.	X			
Hungary	3	2,19	The following should also be taken into consideration in defining Level 2 PSA risk metrics: (a) Current definition of probabilistic safety goals or criteria in use in other Member States; (b) Operating experience feedback; (c) The relationship between defined safety goals related to different PSA levels (e.g. between core damage frequency and large early release frequency); (d) Implications of exceedance of probabilistic safety goals or criteria; (e) Strategies to cope with the exceedance of probabilistic safety goals or criteria. (f) Regional characteristics in terms of population density, distance to major urban areas and routes of isotope transport, economic aspects of the territory, etc.	I think a major reason behind the differences in LRF/LERF definition is due to the differences in the regional characteristics of different countries and even regions. In a low population density area a large release may have much easier challenges in terms of relocation, evacuation, etc. than in highly populated areas. This means that countries and facilities with high population density may develop much stricter definitions and criteria, which should be reflected in the recommendation			X	Siting aspects for the probabilistic criteria are covered by Level 3 PSA. Level 2 PSA criteria is related to the amount of radioactive releases proyected to the environment due to failure of the confirnement function.
ENISS	6	2,21	Later changes can be addressed in the framework of the periodic safety reviews, as part of a living PSA programme, as described in paras. ~~2.20~~2.20–2.23.	Suppression of the repetition.	X			
USA	1	2,21	Suggest adding the following before the last sentence: “In updating a PSA, account should be taken of changes in the design and operation of the plant (operating procedures and practices, emergency operating procedures, maintenance policies, operator training, accident management practices etc.), changes to external facilities or sources of external hazards, new technical information, more sophisticated PSA methods and tools that become available, changes to industry operating experience and new plant specific data derived from the operation of the plant, e.g. data to be used for the assessment of initiating event frequencies or component failure probabilities. “	The text in para 2.21 related to living PSA could be enhanced to explain what type of plant changes should be monitored. While some are mentioned (initiating event frequencies, failure probabilities, PSA methodologies), others are not. In particular, changes to operating and emergency procedures and changes to external hazards could be mentioned.	X			
Germany	10	2,25	Paragraphs 2.25–2.26 provide recommendations on meeting Requirement 1 of GSR Part 4 (Rev.1) [2] on a graded approach and Requirement 14 of GSR Part 4 (Rev.1) [2] relating to the scope of the safety analysis for a PSA.	Please change to “GSR Part 4 (Rev.1)”. Same for paras 11.21 and 12.1.	X			
Hungary	4	2,29	PSA can provide useful insights and inputs for various interested parties, such as operating organizations (management and engineering, operations and maintenance personnel), regulatory bodies, technical support organizations, designers and vendors, for making decisions on: (a) Design modifications and plant modifications; (b) Optimization of plant operation and maintenance; (c) Safety analysis and research programmes; (d) Regulatory issues. (e) Scenarios to be focused on during emergency preparedness drills/training planning (f) SAMG development and operator training optimization	I think these two items should be added to the list as prime examples on the use of level 2 PSA results.	X			
Germany	11	2,31	2.31, Line 6, last sentence: … Later changes can be addressed in the framework of the periodic safety reviews, as part of a living PSA programme, as described in paras~~.~~ ~~2.20~~2.20–2.23.	Typo	X			
Mexico	2	2,31	…paras. 2.20~~2.20~~ 2.23	Delete the duplicate number 2.20	X			

ENISS	7	2,34	The PSA should aim to identify all accident sequences that contribute in a non-negligible way to risk. If the PSA ~~analysis~~ does not address all significant contributions to risk or if its scope is reduced (see para 2.25) (e.g. if it omits external hazards or shutdown states), then the insights ~~conclusions~~ drawn from the PSA about the level of risk from the plant, the balance of the safety features provided and the need for changes to be made to the design or operation to reduce risk might be limited ~~biased.~~ Such limitations should be acknowledged when using PSA to support decision making. Therefore, consistently with the graded approach of the requirement 1 of the GSR Part 4, ~~the use of~~ the ~~full scope~~ PSA model should be full scope ~~is therefore recommended~~ if possible and relevant, or a with a limited scope but with alternative approaches to provide the necessary insights.	Text improvement Proposal to qualify the recommendation because a full scope Level 2 PSA (including in particular a comprehensive list of hazards and combination of hazards) implies significant resources, potentially in contradiction with the graded approach put forward in the requirement 1 of the GSR Part 4.		X The PSA should aim to identify all accident sequences that contribute in a non-negligible way to risk. If the PSA does not address all significant contributions to risk or if its scope is reduced (see para 2.25) (e.g. if it omits external hazards or shutdown states), then the insights from the PSA about the level of risk from the plant, the balance of the risk contributors provided and the need for changes to be made to the design or operation to reduce risk might be limited. Such limitations should be acknowledged when using PSA to support decision making and addressed by alternative analysis when needed.		Term introduced "risk contributors" to be in compliance with the purpose of the sentence.
Japan	8	2,34	The PSA should aim to identify all accident sequences that contribute in a non-negligible way to risk. If the analysis does not address all significant contributions to risk (e.g. if it omits external hazards or shutdown states), then the conclusions drawn from the PSA about the level of risk from the plant, the balance of the safety features provided and the need for changes to be made to the design or operation to reduce risk might be biased. Such limitations should be acknowledged when using PSA to support decision making. ~~The use of the full scope PSA model is therefore recommended.~~	The last sentence may result in the misunderstanding that full scope PRA can diminish the limitation of using PSA. It should be deleted.		X...Therefore, consistently with the graded approach of the Requirement 1 of the GSR Part 4 [2], the PSA model should be full scope if possible and relevant, or a with a limited scope but with alternative approaches to provide the necessary insights.		Text modified to be in compliance with SSG-3
Germany	12	2,35	The results of the PSA should be used to identify weaknesses in the design or operation of the plant as well on actions considered in guidelines for severe accident management ~~guidelines~~ strategies (see also IAEA Safety Standards Series No. SSG-54, Accident Management Programmes for Nuclear Power Plants [14]).	Clarification	X			
China	3	3	Section 3: "OBJECTIVES AND CONTENT OF DOCUMENTATION" in section 12, i.e. paragraphs 12.1 to 12.12, moves to section 3 to generate one subsection "General aspects of Level 2 PSA documentation" at the end of section 3.	In SSG-3 (DS523-Step12), "GENERAL ASPECTS OF PSA DOCUMENTATION" is provided in Section 3. However, some descriptions about "GENERAL ASPECTS OF PSA OCUMENTATION" are provided in Section 12. In Section 3, it is better to provide general document guidance. And in section 12, it is better to provide Level 2 PSA specific document guidance.		X 3.14 (h) (g)(h)Scope and structure of the documentation for the Level 2 PSA (See section 12) General recommendations related to the documentation of PSA results are addressed in paras 3.17 to 3.19 of SSG-3 (Rev. 1) [4] and are not repeated here.		Sentence added in 3.14. Proposed to delete paras from 12.1 to 12.3 and make reference to SSG-3 (Rev.1) since the recommendations in those paras are similar.
Bulgaria	7	3,6	A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs being part of the installation containing potential sources of radioactive releases ~~other nuclear power plants~~	The last part doesn't make any sense.		X… the installation containing other potential sources of radioactive releases …		Better reading
Canada	2	3,6	"A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs being part of the installation containing potential sources of radioactive releases from other nuclear power plants…"	Clarification		X…the installation containing other potential sources of radioactive releases …		Considered togetehr comment 7 Bulgaria
China	4	3,8	The description "paras 3.3–3.14 of SSG-3" modifies to "paras 3.3–3.11 of SSG-3".	Paras 3.12–3.14 in SSG-3 is "Team selection and organization" specified to Level 1 PSA, which is not very suitable to Level 2 PSA. "Team selection and organization" for Level 2 PSA has been provided in paras 3.18-3.21 that is specified to Level 2 PSA, which is enough.	X			
Saudi Arabia	4	3,10	[….] analyses of both the behaviour *of the containment* during the severe accident as well as and the radiological source terms are subject to large uncertainties associated with phenomena.	Better formulation	X			
Germany	13	3,11	In accordance with the requirements established in GSR Part 2 [5], a management system for the project should be implemented with due consideration given to the safety implications of the results of the Level 2 PSA and its intended uses. In particular, the application of expert judgment should be justified and managed through a controlled and documented process. Provisions should be made by the Level 2 PSA project management for establishing independent review processes or performing comparative studies, as appropriate (see paras 3.23- 3.28).	Clarification Please insert "paras" for similar cases also in 3.11, 10.12,11.8, 11.20, 11.22 bullet (d), 14.27 and 15.4	X			
ENISS	5	3,16	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 3.16 …criteria for the credited systems…		In compliance with SSG-3
Finland	2	3,16	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "credited systems"		The term "safety systems" in this context refers to system considered in PSA, so the term is changed to "credited systems" in compliance with SSG-3
Saudi Arabia	5	3,16	[….] such as the thermohydraulic codes used to support the success criteria for safety the credited systems in the Level 1 PSA	Not all the credited systems are safety systems.	X			
China	5	3,18	In the selection of the Level 2 PSA team, it should be ensured that there is an adequate level of expertise in the following areas: (i) knowledge of the design and operation of the plant, (ii) knowledge of severe accident phenomena, and on challenges to the containment, (iii) knowledge of radioactive material source term, and (iii) knowledge of PSA in general, and of Level 2 PSA techniques in particular.	In the engineering practice, the experts who knows severe accident phenomena are familiar with the physical and thermal hydraulic phenomena, but not familiar with radioactive material source term, so the experts focus on this area are recommended.		X… knowledge of physics regarding radioactive material…		The point is to have knowledge on the phenomena related to radioactive material.

Saudi Arabia	6	3,18	[…], and probabilistic safety analysts specialized in severe accident phenomena and other Level 2 PSA disciplines is essential.	In Level 2 PSA, the focus is on severe accident phenomena.	X			
Saudi Arabia	7	3,20	Regarding the knowledge of PSA in general and of Level 2 PSA techniques in particular, the Level 2 PSA team members using computer codes:	Not all the members of the Level 2 PSA are using computer codes.	X			
Saudi Arabia	8	3,21	For a nuclear power plant in operation, the Level 2 PSA team should *include* ~~consider including~~:	More concise formulation.			X	Not all Level 2 PSA teams have the same organization, and the formulation "…should consider including" is more flexible.
Saudi Arabia	9	3,21	Experts in the structural design, the pressure *load bearing* capacity and the failure modes of the containment	More precise formulation		X …load bearing…		Agree but loads are not just from pressure, temperature is also a load.
Saudi Arabia	10	3,21	Experts in developing event tree analysis, fault tree analysis, human reliability analysis, uncertainty analysis, and statistical methods, all in particular for Level 2 PSA;	Better formulation			X	"all" implies that experts should be for Level 2 PSA, and not for Level 1 PSA.
Germany	14	3,23	… This internal independent verification process may help identifying some sources of uncertainties (~~e.g.~~ see e.g. paras 5.13, 6.24-6.27, 7.23-7.30, 8.18-8.22. and 11.22 ~~11.21-(4)~~).	Sources of uncertainties are dealt with in para. 11.22, please verify.		X…(e.g. see paras 5.13, 6.24-6.27, 7.23-7.30, 8.18-8.22. and 11.22 11.21-11.25(4)).		The description of sources of uncertainties starts in 11.21 with reference to tables 4 and 9.
Germany	15	3,24	Since the development of the Level 2 PSA and the design may be conducted in parallel as part of the iterative design process of the nuclear power plant, the licensee organization should carry out an independent verification (e.g. peer review) to ensure that Level 2 PSA results only relate to the design and operation of the nuclear power plant as submitted to the regulatory authority for approval (i.e. ~~according to~~ in accordance with the scope of the document to be submitted to the regulatory authority), and comply with relevant regulatory requirements related to reference values and risk metrics for Level 2 PSA.	Clarification, as phrases “in accordance with” and “according to” do not have the same meaning. Please verify. Same for paras 7.8 and 10.1 (last sentence).	X			
Saudi Arabia	11	3,24	Since the development of the Level 2 PSA and the design *of the nuclear power plant* may be conducted in parallel as part of the its design process, the licensee organization should carry out an independent verification […]	Better formulation	X			
Saudi Arabia	12	3,24	[….] regulatory ~~authority~~ *body* for approval…	Please consider consistently using either regulatory body or regulatory authority throughout the whole safety guide.	X			Text modified accordingly in the whole document
Saudi Arabia	13	3,25	The independent verification of the Level 2 PSA performed by or on behalf of, the licensee organization should be conducted by a different group of experts or institution from those who develop the Level 2 PSA (e.g. external group or institution of the licensee organization, sometimes from a different Stat), to ensure that the Level 2 PSA conforms to current, internationally recognized good practices in Level 2 PSA).	Editorial (closure parenthesis was misplaced).	X			
Germany	16	3,28	The report compiling the results of the independent verification of the Level 2 PSA should consider the assessment of the appropriateness and comprehensiveness of: …. (h) Structural analysis and/or fragility curves;…	Clarification	X			
Hungary	5	3,28	The report compiling the results of the independent verification of the Level 2 PSA should consider the assessment of the appropriateness and comprehensiveness of: (a) The PDSs development, grouping and quantification (if the Level 1 and 2 PSA models are not integrated)/The connection of the level 1 and 2 PSA models (if the models are integrated); (b) The analysis of accident progression and the associated systems; (c) The models of phenomena that could occur in relation to the behaviour of the containment of the nuclear power plant following core damage; (d) The accident progression event tree models and supporting models as well as the methods for solution of the logic models; (e) The probability development (e.g. phenomena probabilities based on data or expert judgement); (f) The release categories development, grouping, quantification and source term characterization; (g) Supporting calculations, correct and appropriate application of codes; (h) Structural analysis/fragility curves; (i) The models for considering the human reliability analysis; (j) The consideration of equipment reliability taking into account the equipment qualification or survivability (in particular for severe accidents scenarios); (k) The uncertainties and sensitivity analysis carried out (e.g. the bases for the selection of probability distributions of uncertain parameters and assignment of their distributions parameters).	Recommendations should be provided for both approaches and it should be highlighted that the PDS appropriateness checking is only a step if the models are not integrated.			X	Independently of the approach followed (i.e. integrated of separated) the Level 2 PSA model starts with the development of PDS resulting from the list of internal events (see para Section 5).
Saudi Arabia	14	3,28	The uncertainties and sensitivity analysis carried out (e.g. the bases for the selection of probability distributions of uncertain parameters and assignment of their distributions parameters)	See comment on uncertainty evaluation below. One of the most difficult steps in uncertainty evaluation is the proper selection of the probability distributions of uncertain parameters.	X			The recommendation is related to those aspects that need to be presented for the independent review. Conducting uncertainty and sensitivity analyses are an important part of the Level 2 PSA, which include deterministic and probabilistic aspects, and the results of the uncertainty and sensitivity analyses should be made available for the independent review.
Canada	3	4,1	“Design features that can influence the progression of a severe accident and Level 2 PSA include: fan coolers, containment sprays and/or filtered containment venting systems, ~~and~~ suppression pools and hydrogen control systems (ignitors, recombiners).”	Hydrogen control systems will influence the progression of the severe accident.	X			

China	6	4,1	Design features that can influence the progression of a severe accident and Level 2 PSA include: Fan or water-cooled Cooler, containment spray and/or filter containment exhaust system and suppression tank, primary circuit pressure relief system, Core melt In-vessel Retention system, etc.	Add some systems that influence the progression of a severe accident.		X...Design features that can influence the progression of a severe accident and Level 2 PSA are reactor technology and design dependent includesuch as: fan or water-cooled coolers, heat removal, containment sprays and/or filtered containment venting systems, and containment exhaust system and suppression pools, dedicated set of steam relief valves, and hydrogen control systems (i.e. ignitors, recombiners).		Modification to be more general and in accordance with terminology of SSG-56. Changing the term "include" to "such as" does not need to be comprehensive.
France	3	4,1	Design features that can influence the progression of a severe accident and Level 2 PSA include: fan coolers, containment sprays, heat removal, and/or filtered containment venting, systems and suppression pools »	Completeness of the example		X… containment heat removal system		In compliance with the terminology in SSG-53
Hungary	6	4,1	... Before starting the analysis, the Level 2 PSA team ~~should~~ has to become familiar with the design, ~~and~~ operation of the plant. The aim should be to identify and highlight plant SSCs and operating procedures...and suppression pools. ~~This exercise should include the reactor building and/or the auxiliary building and the secondary containment or other relevant structures and buildings which all depend on the reactor technology and design.~~ For existing plants, familiarization with the plant should include ~~a~~ :	While SSGs are supposed to provide recommendations, I believe that stating to familiarize with the design and operation with the facility before developing the PSA models is a triviality and should be stated as such.			X	It might seems trvial recommending to familiarize with the plant design, but it is needed since it is an important part of the methdology. The part proposed to be deleted is important and necessary.
Hungary	6	4,1	plant walk~~-through~~downs ~~and should involve~~with the participation of operating personnel and plant technical staff.	I also believe that the deleted part is unnecessary and also incomplete, since it doesn't include parts like the annulus. I think this list can be developed based on the description in the previous text and as stated it is design specific.	X			The annulus is part of the reactor containment building (see SSG-53, para 5.13). The text leave the option open to different reactor technologies and designs
Hungary	6	4,1	Interviews with the staff fulfilling relevant roles from a level 2 PSA point of view The plant familiarization should involve all members of the Level 2 PSA team.	I also suggest to add the interviews to the list for what should be included in the plant familiarization.		X...Interviews with operating personnel and plant technical staff fulfilling relevant roles from a Level 2 PSInterviews with operating personnel and plant technical staff fulfilling relevant roles from a Level 2 PSA perspective should also be conducted.		Modification for better reading as an additional recommendation.
Saudi Arabia	15	4,1	For existing plants, familiarization with the plant should include a plant walk-through walkdown and should involve the participation of operating personnel and plant technical staff.	For consistency.	X			
France	4	4,3	Table I: Water volume available for containment pressure control or fission product retention ~~and atmosphere volumes~~	More precise on the role of suppression pool volume in severe accident conditions.	X			
France	5	4,3	Table I: Design of some ~~Generation III+~~ plants (recent or backfitted) includes some features for cooling of the spread molten core	This feature is now applicable to some Gen II upgraded NPPs.	X			
Russian Federation	2	4,3	Table I: Full inventory of radionuclides in the core for the end of the nuclear fuel cycle of a stationary fuel load.	In order to take into account the worst consequences, it is proposed to add "for end of the nuclear fuel cycle ofa stationary fuel load.			X	To avoid being overly prescriptive.
Saudi Arabia	16	4,3	Table 1: […], nominal and maximum temperature of ~~flow rate~~ *the coolant*	Better formulation	X			
Saudi Arabia	16	4,3	Table 1: ~~Safety systems~~ Systems actuation mechanism	Not all credited systems are safety systems	X			
Saudi Arabia	18	4,11	During the progression of a severe accident *involving the degradation* of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors)	Better formulation as "severe accident of the fuel" is not appropriate.	X			
Finland	3	4,13		Footnote 9 is missing			X	Paragraph deleted since it could be considered as a duplication of para 4.10. See comment 17 Saudi Arabia
Germany	17	4,13	Paragraphs ~~0~~ 4.14-4.15 provide recommendations on relevant information on safety provisions …	Please replace placeholder '0' for paragraph number 4.14.			X	Paragraph deleted since it could be considered as a duplication of para 4.10. See comment 17 Saudi Arabia
Israel	7	4,13	The digit 0 (zero) has to be replaced in the paragraphs referred to as 0-4.15				X	Para deleted since the recommendation is repeated see para 4.10
Saudi Arabia	19	4,13	~~Paragraphs 0-4.15 provide recommendations on relevant information on safety provisions 9 that should be collected in the familiarization task for the Level 2 PSA related to the success of strategies to deal with core damage10.~~ ~~Note: Footnote 9 is not defined, and footnote 10 needs to be introduced in page 3, para. 2.9 where "core damage" appears for the 1st time.~~	This paragraph is to be removed because it is redundant with para. 4.10, which should be modified to read as: Paragraphs 4.11 to 4.14 aim at providing an overview of key aspects to be considered from the plant familiarization perspective in relation to strategies to cope with severe accident associated phenomena when performing a Level 2 PSA. Of course, the numbering of the paragraphs that follow needs to be changed.	X			
Bulgaria	8	4,14		Only ex-vessel IVR is discussed. The text should be expanded with aspects related to IVR by in-vessel injection (e.g. TMI-2)			X	Para 4.14 (now 4.13) describes both in-vessel and ex-vessel injection (see 4.13 main text and (b)).
Bulgaria	10	4,14	(f) Water inventory available (i.e. affecting the delay the time of corium arrival in the lower plenum and therefore reduce the residual heat removal~~power to extract~~	Residual heat removal is more often used in the literature.	X			
Bulgaria	9	4,14	footnote 9 Corium is a complicated mixture of fuel, zirconium alloy and steel, which forms in water cooled reactors as a result of thermochemical reactions~~, including between zirconium and water~~ .	It is not clear why reaction between Zr and water needs to be emphasized when a corium definition is provided. What is so special about it? This reaction is more relevant when hydrogen or heat generation are discussed.		X Footnote 9: Corium is the material formed during the meltdown of a nuclear reactor. It is composed of nuclear fuel (uranium or plutonium) and material that melts on contact with the fuel. Ref. http://www.futura-sciences.us/dico/d/chemistry-corium-50003906/		Definition of corium as provided in a reference.

Finland	4	4,14	4,.14 replace whole footnote 11 with: "Focusing effect" phenomenon involves a thin metal layer on top of a large oxidic pool. If radiative heat transfer on top is inadequate to discharge the thermal power received from below, the temperature of the metal layer would increase, and increasing amounts of energy flow would be directed to the vessel wall. This focusing would increase as the metal layer thickness decreases, until the thermally-induced turbulence in it is sufficiently low for it to sustain significant radial thermal gradients.	The current definition is unclear. We propose to use the definition by Theofanous in T.G. Theofanous: "In-vessel retention as a severe accident management strategy" (https://inis.iaea.org/collection/NCLCollectionStore/_Public/44/026/44026286.pdf)		X… [Definition added as proposed in the reference.]This effect can induce reactor vessel rupture.		The definition was improved with the proposed text in the reference.The reference was also added. In addition, a sentence at the end was added to consider the effects of this focusing effect on the reactor vessel, which is not presented in the referred definition.
Pakistan	3	4,14	...the integrity of the reactor pressure vessel by cooling it from outside and the ~~integrity~~**cooling** of the corium inside by in-vessel water.	The corium is produced after loss of integrity of reactor core; therefore, the term "integrity" may be replaced with "cooling"	X			
Saudi Arabia	20	4,14	Design solutions related to reactor pressure vessels internals (e.g. large mass of steel *in the corium relocated in* the lower plenum may reduce the risk of the focusing effect at the reactor pressure vessel wall)	More precise formulation	X			
Saudi Arabia	21	4,14	Water inventory available (i.e. affecting the *time* delay ~~the time~~ of corium arrival in the lower plenum and therefore reduce the residual power to extract).	Better formulation	X			
Israel	8	4,15	Page 18 footnotes: It seems that footnote 9 is missing or the number 9 was skipped during numbering the footnotes			X Footnotes numbers will be updated in the later phase of technical editing.		
Saudi Arabia	22	4,16	The data necessary for the *Level 2* ________________________________ PSA depend in part on the scope of the analyses and the nature of the computational tools.	More precise formulation	X			
Germany	18	4,18	… However, great care ~~has to~~ should be applied when drawing conclusions from such a comparison.	Clarification, as recommendations in IAEA Safety Guides are 'should' statements.	X			
Israel	3	4,18	Regarding reference plant in the development of Level 2 PSA, it is should understood that in fact similar plants are (of course) not identical…Therefore we suggest to point out at this place in the paragraph the warning written at the end of the paragraph ("However great care has to be applied when drawing conclusions from such a comparison"), emphasizing the difficulties in scaling between plants – considering in many cases the scaling is not necessarily linear.	Completeness	X			
Hungary	7	5,1	This section provides recommendations on the interface between Level 1 PSA and Level 2 PSA in case the Level 1 and 2 PSA models are not integrated. It addresses the analysis of results and information from the Level 1 PSA that need to be carried out to provide the necessary input for the Level 2 PSA. The detailed implementation of Level 1 PSA and Level 2 PSA interface will depend on methodology chosen for the Level 2 PSA, the modelling software used and the reactor technology.	It should be specified that the recommendations in Chapter 5 refer to non-integrated PSA models, since most of these recommendations cannot be followed in case of integrated models.			X	Independently of the approach followed (i.e. integrated of separated) the Level 2 PSA model starts with the development of PDS resulting from the list of internal events (see para Section 5).
Bulgaria	11	5,2	PDSs ~~should represent groups of accident sequences that have similar accident timelines, containment status and containment system (un)availability status and which generate similar loads on the containment, thereby resulting in a similar event progression and similar radiological source terms.~~ a~~A~~ttributes of accident progression that will influence the chronology of the accident, the progression of the core damage, the containment response or the release of radioactive material to the environment should be identified. The attributes of the PDSs provide boundary conditions for the performance of severe accident analysis.	The two sentences actually cover the same information.			X	The first sentence provides recommendations related to the grouping of PDS, while the second is related to the attributes needed to be considered for that purpose. Of course they are very much interrrelated but the recommendations are not the same.
Saudi Arabia	23	5,3	Proper care should be taken to ensure that ~~optimisms~~ *deviations, which minimize the source term* , are not introduced when sequences from the Level 1 PSA are mapped and transferred to the Level 2 PSA and that no sequences are lost or duplicated.	More precise formulation		X…that ~~optimisms~~ assumptions, which could minimize the source term,…		Consider "assumptions" is a better term. Also "could" is added since some assumptions might not impact the source term.
UK	1	5,3	New paragraph suggested between 5.3 and 5.4 Suggested paragraph: "The grouping of Level 1 PSA sequences into PDSs often requires some assumptions and simplifications to be applied which may introduce additional uncertainties. Special care should be taken to keep track of any assumptions and simplifications so that the additional uncertainties are not overlooked during the uncertainty analysis."	The current version does not instruct the analyst explicitly to keep track of any assumptions and simplifications so that the additional uncertainties are not overlooked during the uncertainty analysis.	X			
China	7	5,4	The description "the safe state" modifies to "the stable controlled state".	Following severe accident, the core has damaged. The aim for actuation of mitigation measures is to make the molten core stable controlled, which is not the safe state usually mentioned in deterministic analyses.		X...the system mission time to reach a controlled state or to fulfil the modelled system function		To be in compliance with terminology used in SSR-2/1 (Rev. 1) Requirement 20: "controlled state".
China	8	5,4		Providing the example for the mission time, which is better in Level 2 PSA guidance.			X	Examples of mission time are plant and technology specific, so difficult to reach consensus.
Saudi Arabia	24	5,4	The success criteria for system modelling in PSA should specify the system mission time ~~to reach the safe state~~ or to fulfil the modelled system function. In particular for Level 2 PSA, the mission time should be defined adequately for capturing the severe accident progression, the time needed for design features to effectively cope with severe accidents, including possible cliff-edge effects, and to ensure that the residual risk accrued after the mission time is negligible.	There is no consensus on the determination of a safe state for severe accidents.		X ...5.4. The success criteria for system credited modelling in Level 2 PSA should specify the system mission time to reach ~~the safe~~a controlled state or to fulfil the modelled system function. ~~In particular for Level 2 PSA, t~~The mission time should ...		To be in compliance with terminology used in SSR-2/1 (Rev. 1) Requirement 20.
Finland	5	5,5	Table 3: Reactor coolant system coolant inventory (shutdown states): — Full power inventory — Flooded refuel cavity — Mid-loop operations in a pressurized water reactor	Just to clarify that the coolant inventory is meant, not fission product inventory.	X			

Bulgaria	12	5,5	Table 3: State of fuel in the reactor for decay heat (shutdown states): — Pre/post refuelling — ~~Time since reactor shutdown~~	How these two states can be mutually exclusive in order to be used for the attribute in question? The selected states of the attribute should be mutually exclusive.		X… State of fuel in the reactor for decay heat: —Operating power level —Pre/post refuelling —Time since reactor shutdown		To be more general with regard to all decay heat to be removed. See comment 6 from Finland.
Finland	6	5,5	Table 3: State of fuel in the reactor for decay heat ~~(shutdown states):~~ — Operating power level —Pre/post refuelling — Time since reactor shutdown	Plants may operate in partial power (load following) depending on electricity price and electricity need. We have seen that electricity price has become negative more frequently in recent years and plants have reduced power output. Operating the plant in partial power may be done also from other reasons like environmental permit limitations. This kind of operation should be considered if it is done at the plant in question. Decay power level has impact on core melting, hydrogen generation etc. Also reference to shutdown states can be removed as it is unnecessary.	X			
Bulgaria	13	5,5	Table 3: Status of emergency cooling system and other cooling systems ~~(timing of core damage and ability to prevent further core damage progression)~~ Timeing of core damage (This is related to the onset of release. It is important that early and late release sequences can be identified in the APET. It is also important to consider nominal leak contributions when describing the release categories, see chapter 7.)	First, it is not clear why ECCS is interfered with core damage time. Second, the attribute (time of core damage) should be considered separately, since it helps easily to distinguish between fast and slow running accident progressions for which the emergency planning response could be different. This attribute is also presented as a separate one in “Table 1 Example of Plant Damage State Attributes” of ASAMPSA2.			X	ECCS and other cooling systems have an important role in arresting the accident progression to severe accident (i.e. core damage). "Timing" is the appropriate term since it defines the time when something happens.
Bulgaria	14	5,5	Table 3: Status of containment’s engineered safety features (Status of hydrogen ~~control~~ management systems, i.e., monitor and control) ***Comment: Add information about the PARs and possible states of the attribute in question. For example:*** ***- Fully available as designed*** ***- Fully failed*** ***- Degraded***	It is not clear whether hydrogen control systems attribute consider PAR operation or hydrogen measurement only. If so, corresponding states of the attribute need to be presented as an example.		X… Status of means for management of combustible gases, i.e. monitor and control (e.g. passive means (e.g. autocatalytic recombiners) and active means (e.g. igniters)) — Fully available as designed — Degraded — Fully failed		Text modified to be in accordance to terminology used in SSG-53.
Bulgaria	15	5,5	Table 3: Containment status ***Comment: State "bypassed is missing".***	This state is discussed in para. 5.8, and it should be included here as well.	X			
China	9	5,6	One method for incorporating such missing systems into the Level 1 PSA is to develop extended event trees that link to Level 2 PSA system models, as shown in FIG. 1.	In FIG.1 of this revision, extended event trees is used, not bridge trees.		X...models, presented as extended event trees in FIG. 1…		Text modified to be in accordance with Fig. 1.
Finland	7	5,6	5.6 title: “PLANT DAMAGE STATES FOR PSA FOR INTERNAL INITIATING EVENTS DURING FULL POWER CONDITIONS” change to PLANT DAMAGE STATES FOR PSA FOR INTERNAL INITIATING EVENTS DURING FULL AND PARTIAL POWER CONDITIONS	Operation in partial power should be considered if it is done at the plant in question (e.g. due to repairs or load following operation). The plant status is similar to full power operation except the reactor power is lower. Therefore, the paragraphs are valid also for this kind of operation.			X	This operational mode is covered in the section "PLANT DAMAGE STATES FOR LOW POWER AND SHUTDOWN MODES OF OPERATION"
Finland	8	5,6	For example, the Level 1 PSA may not have addressed the status of containment systems or other systems that do not directly affect the determination of core damage (i.e. they do not contribute to the success criteria for preventing core damage).	Typo			X	It is not the "termination" but the "determination" of the those systems contributing to core damage.
Japan	9	5,6	If the Level 2 PSA is developed following a separated approach (see para 2.6) Level 1 PSA does not account for specific aspects relevant to the specification of PDSs. For example, the Level 1 PSA may not have addressed the status of containment systems or other systems that do not directly affect the determination of core damage (i.e. they do not contribute to the success criteria for preventing core damage). In such cases, the Level 1 PSA should be expanded to take into account the missing aspects in the specification of PDSs (see Table 3 for reference). One method for incorporating such missing systems into the Level 1 PSA is to develop extended Level 1 event ~~bridge~~ trees that link to Level 2 PSA system models, as shown in FIG. 1, thereby capturing important dependencies (support systems, operator performance, etc.).	To be consistent with FIG.1.			X	The bridge event trees include those extended event trees.
Bulgaria	16	5,8	For PDSs where containment is failed, not-isolated or bypassed, only a source term analysis may be necessary, though a simplified event tree may need to be provided in the model.	**Not isolated** containment can also join the group of sequences for which only a source term analysis can be performed.		X…not isolated		Better reading.
Bulgaria	17	5,9	~~If the~~In any case ~~Level 2 PSA is developed as an extension of Level 1 PSA,~~ the definition and selection of characteristics specified for the PDSs should be justified.	The statement is valid in all cases, i.e., it is not important what approach has been followed.		X...In either approach (see para 2.6), the definition and selection of characteristics specified for the PDSs should be justified.		To provide the link to the relevant para.
Saudi Arabia	25	5,9	It should be noted that the level of detail of characteristics used to define the PDSs depends on the approach ~~case~~ used for the development of Level 1 PSA and Level 2 PSA (see para 2.6).	Better formulation.	X			

Bulgaria	18	5,10	(a) ~~Type of initiating event, which can, for example, affect the rate of discharge of reactor coolant in the containment, the progression of the core damage and of hydrogen generation, and the timing of the release of radioactive material~~	It is clear that what is provided here are examples. Anyway, the role of IE as a PDS attribute is suspicious. Let say the IE is LOOP and during the DBA progression phase, PORV opens and stuck open. How this sequence should be considered? As LOCA or as Transient? The main goal of IE consideration is to delineate between bypass and other sequences. In our practice, we use pressure for not bypassed events (bullet (e) in this list), which gives more valuable information about the accident progression tendency and expected subsequent phenomena			X	The consideration of the type of IEs as an attribute in the PDS grouping not only to differentiate between containment bypass and other sequences.
Bulgaria	19	5,10	(f) ~~The pressure in the reactor pressure vessel at the time of lower head failure may affect the mode of discharge of debris to the containment. This, in turn, could present a challenge to containment integrity if, for instance, high pressure melt ejection and direct containment heating ensue~~	The pressure at VB is more relevant to APET (CET) model and not here. The pressure at the moment of CD is more appropriate as PDS attribute. Moreover, it is not clear how this will be determined before accident progression analysis (deterministic) are not conducted.		X… (e) The reactor coolant system pressure at the onset of core damage and the status of safety valves or relief valves and other components that could change the pressure in the reactor pressure vessel before failure of the lower head of the reactor pressure vessel . (g)a The pressure in the reactor pressure vessel after the onset of core damage also affects the possibility of temperature and pressure induced failures of the reactor coolant system (e.g. creep rupture of piping and steam generator tubes, or thermal seizure of a safety or relief valve in the open position). The pressure will be affected by the initiating event and the functionality of any depressurization system. (f)b The pressure in the reactor pressure vessel at the time of lower head failure is related to (but not necessarily the same as) the pressure at the onset of core damage and may affect the mode of discharge of debris to the containment. This, in turn, could present a challenge to containment integrity if, for instance, high pressure melt ejection and direct containment heating ensue.		The attribute "pressure in the reactor pressure vessel" helps to define the sequences in the development of the APET leading to containment failure and subsequentially to potential radioactive releases. The items f and g were modified as sub-items of (e) and former (f) was modified to be in accordance with the attribute "pressure at the onset of core damage".
Bulgaria	20	5,10	(k) The availability of the containment ~~protection~~ systems	**Protection** is excessive. Without it, the statement is more general.		X...(k) The availability of the containment associated systems		In accordance with the terminology in SSG-53.
Bulgaria	21	5,10	(l) ~~Initial and boundary conditions including the a~~Availability of alternating current and direct current power and associated recovery times.	It is not clear what is meant by "Initial and boundary conditions including the". Availability of power supply is important to delineate sequences with SBO from others where power supply is available. In case of SBO sequence a recovery of power supply can be feasible to be considered in APET or/and in PDS bridge trees.			X	Initial and boundary conditions refer to assumptions considered for the modelling of the progression of severe accident, such as temperature, heat flux, pressure, mass, filter capacity, recombination (PAR) capacity, capacity of DC, etc.
Bulgaria	22	5,10	(m) ~~The actions by operating personnel that have been attempted and failed.~~	This is not consistent with the text in the para. 5.10 where is stated that "account should be taken of the equipment and system failures".		X… 5.10. (new 5.11.) In specifying PDSs without containment bypass, account should be taken of the equipment and system failures or scenario features, including operator actions, identified within Level 1 PSA that could affect either the challenge to the containment or the release of radioactive material.		Text 5.10 (new 5.11) modified to consider operator actions.
Finland	9	5,10		This is very much repetition from TABLE 3. Paragraph 5.10 should be removed. If need be, some of the bullets may be included in TABLE 3.			X	There is a need to repeat recommendations since they are different operational modes.
France	6	5,10	In specifying PDSs without containment bypass, account should be taken of the equipment and system failures or scenario features identified within Level 1 PSA that could affect either the challenge to the containment or the release of radioactive material.	The list introduced by this sentence include information like time, pressure …	X			
UK	2	5,10	Suggested footnote: "In some cases, it is useful to group separately the sequences with high pressure at the onset of core damage but where the pressure is naturally reducing (i.e. due to loss of coolant) and will result in low RCS pressure before the bottom head failure. These sequences should be grouped with the low pressure sequences in order to avoid overestimating the high pressure PDS frequency."	When grouping the PDSs by RCS pressure it is useful to separate the sequences where the pressure is high at the onset of core damage but us gradually reducing (e.g. due to loss of coolant) and will be reduced naturally below HPME range at the time of bottom head failure without any depressurisation measures. These sequences should be grouped with the low pressure sequences in order to avoid overestimating the high pressure PDS frequency.		X… In some cases, it is useful to group separately the sequences with high pressure at the onset of core damage but where the pressure is naturally reducing (e.g. loss of coolant accidents or small consequential leakages) and can result in low RCS pressure before the bottom head failure.		To consider those situations where depressurization may occur as a consequence of the severe accident.

Bulgaria	23	5,11	~~Other attributes of PDSs may be important in some applications of PSA. For instance, if the PSA is being used to help determine accident management measures, then the status of the electrical power supply should be taken into account, since this information may be needed for some later actions.~~	The two sentences implies that AC power attribute should be used as an attribute when PSA applications are developed only. This is not fully correct. The power supply availability is usually part of the main list of attributes.		X… For some reactor designs, the status of electrical power supply is important in grouping of accident sequences into PDS because it influences the application of recoverability of accident management systems.		Text modified to explain the need to additionally consider AC power supply.
Bulgaria	24	5,13	The analyst should justify that the screening carried out does not introduce a significant underestimation of the risk calculated by the Level 2 PSA; careful evaluation is necessary prior to introducing a frequency screening criterion at the PDS level. Note that for some applications screening out approach could be inappropriate, e.g. risk monitor.	In addition to this, it should be mentioned that the screening out approach could also be inappropriate if some applications are foreseen, e.g., risk monitor.	X			Moved to the end of the para for better reading.
Israel	4	5,13	Regarding the approach of using frequency cut-offs, the paragraph does include the warning of "careful evaluation is necessary prior to introducing a frequency screening criterion as the PDS level. Nevertheless, we suggest to include already in this paragraph the text from the later paragraph (par. 5.23): Hazards of very low frequency but with potentially severe consequences…should be considered for the purposes of a Level 2 PSA).	Completeness	X			
China	10	5,15	(a) Decay heat level (time since shutdown from power operation); (b) State of the containment – especially when it is open and associated manual actions to close it prior to core damage; (c) Conditions that determine the time to restore the isolation of the containment and its potentially reduced effectiveness (leaktightness) during such time; (d) The integrity of the reactor coolant system pressure boundary with reactor vessel head removed, nozzle dams installed, safety valves removed, reactor coolant system vent opened; (e) The coolant inventory in the reactor coolant system.	Item (b) and (f) both focus on state of the containment, we suggest to merge them.	X			Item b deleted since item f (updated as item e) could cover both the containment status and the operator actions needed to close it.
Finland	10	5,15	…(b) State of the containment and actions needed to isolate it (including time to perform the actions) (c) [removed] (f) [removed]	Bullets b, c and f consider more or less the same topic. They should be combined.		X...(e)Closure status of the containment and associated manual actions to close it prior to core damage.		Text modification proposed to consider the operator actions. Item b deleted since item f (updated as item e)could cover both the containment status and the operator actions needed to close it. Item c refers to the conditions (e.g. availability of electrical power supply, compressed air, etc) to close the containment and ensure leaktightness.
Bulgaria	25	5,16	In order to extend the scope of Level 2 PSA to include internal and external hazards such as fire, seismic and ~~external~~ flooding...	Flooding could be internal and external, therefore the term **external** next to flooding is misleading.		X...such as fire, seismic hazards and external flooding, …		Text modified for the term "seismic hazards". External flooding was an example, that is why the term "such as…"
Bulgaria	26	5,16	…, should be taken into account (see Ref. [10]), if those aspects have not yet been taken into account in the Level 1 PSA ~~output~~	The word "output" seems to be excessive. Without it, the statement sound more general as it should be. This is also mentioned in 5.17.	X			
Bulgaria	27	5,16	for instance, some failures (e.g., …..) of the systems ensuring the confinement function could be assimilated into already defined isolation failures for systems ensuring the confinement function.	The statement is not fully clear. What is meant by some failures? Failures due to external hazard? Please, give some example.		X text added refering to internal and external hazards		Clarification provided.
Germany	19	5,16	In order to extend the scope of Level 2 PSA to include internal and external hazards such as fire, seismic hazards and external flooding, the potential impact of the hazards on systems necessary for mitigation of severe accidents,...	Please insert "hazards" or, alternative, "events" by **seismic**.	X			
Bulgaria	28	5,18	Depending on the analyst choice, human actions that occur before or soon after core damage may be credited in the Level 1 PSA and captured as part of an attribute to the PDSs for the Level 2 PSA as described in para. 8.4. The way, these human actions are included in the L1 PSA model should not deteriorate L1 PSA results (CDF/FDF).	The statement is not fully clear. If these actions cannot prevent CD, then including them into L1 PSA will overestimate CDF. Moreover, item 8.4 is related to reassessment of operator action in order to lower the conservatism and to capture any specifics of L2 PSA.			X	The recommendation proposed is more pertinent for the Level 1 PSA.
China	11	5,19	"earthquake resulting in a station blackout and a containment failure, perhaps with consequential internal fire or flooding"	The IAEA TECDOC-1791 document states that the design extension condition do not include external hazards, it is recommended to change " A design extension condition earthquake" to "earthquake".		X...~~—A~~ beyond design basis earthquake resulting in a station blackout and a containment failure, perhaps with consequential internal fire or flooding;		Terminology used to be in compliance with IAEA SSG-67.
Saudi Arabia	26	5,19	A ~~design extension condition~~ *An earthquake of a level more severe than the one considered for design* resulting in a station blackout and a containment failure, perhaps with consequential internal fire or flooding;	The terminology "design extension earthquake" does not exist in the IAEA Safety glossary.		X ...~~—A design extension condition earthquake~~A beyond design basis earthquake resulting...		Terminology used to be in compliance with IAEA SSG-67.

USA	2	5,20	Propose delete paragraph 5.20.	Propose delete this paragraph. The information in para 5.20 appears to be a repeat of para 1.18 and 15.3. It does not appear to contain any information related to Section 5 which is dedicated to the interface between Level 1 and Level 2 PSA. The scope of the PSA is sufficiently elaborated in para 1.18, 2.7, 2.25, 2.34 and particularly para 15.3. Furthermore, this paragraph appears to require a full scope Level 2 PSA, which is in contradiction with the other paras on scope of the PSA.		X 5.21. In order to be widely applicable, the Level 2 PSA for hazards should be based on a Level 1 PSA covering these hazards as described in SSG-3 (Rev. 1) [4].		Para 5.20 recommends the recommended scope of Level 1 PSA for the development of Level 2 PSA for hazards. Text modified to be in accordance with the scope of Level 1 PSA for hazards as defined in SSG-3 (Rev.1). Other paras refers to different topics as: Para 1.18 defines the scope of recommendations provided in this safety guide for the development of Level 2 PSA Para 2.7 provides general recommendations related to the objectives of Level 2 PSA. Para 2.25 provides recommendations related to the comparison of results of PSA with defined goals. Para 2.34 provides recommendations related to the graded approach and alternative approaches when a limited scope is considered. Para 15.3 provides recommendations related to the use of Level 2 PSA for introduction of the following paras.
Saudi Arabia	27	5,21	It should be noted that the development of a Level 2 PSA for hazards depends on the scope set for the Level 2 PSA but can also be influenced by the ~~L1~~ *Level 1* hazards PSA results	For consistency.	X			
Germany	20	5,23	Those hazards, single as well as combined ones, which were screened out from further (bounding or detailed) analysis within the Level 1 PSA should also be reassessed, consistent with paras. 6.17 - 6.19 of SSG-3 (Rev. 1) [4] ~~paras. 6.17 to 6.19~~, noting that the latter explicitly states ….	Editorial	X			
Bulgaria	29	6,1	Subsequently, analyses would be performed to support the definition of PDSs, assisting the event tree analysts in establishing which systems and accident progression features are most important for determining the plant response ~~and hence, needing to be included in the PDS definitions~~.	The statement introduces limitation of the analyst freedom. It depends on analyst approach where to account for some systems, i.e. in APET or PDS.		X ...needing to be included in the PDS definitions or in the APET model, as appropriate for the methods being applied in the PSA.		Text modified to provide the choice for considering where the systems could be included in the PDS or the APET.
ENISS	8	6,1	A third area where severe accident analyses provide support is in the assessment of specific phenomena, where accident analysis results may be an input to phenomenological probability calculations (see Section ~~10~~9), and support in defining timing for human action events included in the logic models. Finally, the severe accident analyses support the grouping of accident progression event tree sequences into release categories (see Section ~~0~~10), a similar process to the definition of PDSs, and are also performed to generate the quantitative characterization of radiological release associated with each release category.	Reference to Section 9 seems more appropriate. Section 0 does not exist. Reference to Section 10 seems appropriate.	X			
France	7	6,1	section 10	Typo correction	X			
Germany	21	6,1	The severe accident analysis task typically consists of different groups of analyses, performed in different phases of a Level 2 PSA project. Early on in the project, severe accident analysis would be used to understand the general ~~post-~~core damage accident progression for key initiating events, providing a starting point for the Level 2 PSA analysis (see Ref. [15]).	Clarification: “post-core” is not an expression commonly use.	X			
Germany	22	6,1	… For example, investigations may provide insights on the variation of accident progression when different numbers of injection trains are operating or provide insights into the impact of primary and secondary pressure status (in a pressurized water reactor) or indicate the effect of changes in the volume of water injected to containment on ex-vessel molten core behaviour ~~ex-vessel.~~	Clarification	X			
Germany	23	6,1	… Finally, the severe accident analyses support the grouping of accident progression event tree sequences into release categories (see Section ~~0~~9), a similar process to the definition of PDSs, and are also performed to generate the quantitative characterization of radiological release associated with each release category.	Editorial. Reference should be “Section 9”	X			
Israel	9	6,1	In the two before the last line of this paragraph, referring to section 0, the zero has to be changed		X			
Saudi Arabia	28	6,1	Finally, the severe accident analyses support the grouping of accident progression event tree sequences into release categories (see Section 1 ~~0~~), a similar , a similar process to the definition of PDSs, and are also performed to generate the quantitative characterization of radiological release associated with each release category.	Section 0 does not exist and should be changed to the relevant section, as applicable.	X			Link to section 10 added.
Bulgaria	30	6,2	~~can be joined to~~ support source term analysis (see Section 10)	Better to use "support" instead of can be joined.	X			
Germany	24	6,5	Severe accident progression analysis should be performed by teams with experience in application of severe accident codes; if not, training ~~has to~~ should be included in the project (see paras. 3.18-3.21 on team selection for the Level 2 PSA project).	Please change to “should” statement	X			

Saudi Arabia	29	6,5	Severe accident progression analysis should be performed by teams with experience in application of severe accident codes; if not, appropriate training has to be included in the project (see paras. 3.18-3.21 on team selection for the Level 2 PSA project).	We should not give the impression that a light training is enough to make staff capable of performing reliable calculations. As recommended in para. 3.20, the teams should be able not only to use the computer codes but also to understand their limitations and to soundly interpret the results of calculations.	X			
Finland	11	6,7	The analysis of the progression of the severe accidents in the reactor, which were identified by the Level 1 PSA and grouped in specific PDSs, should provide key information such as fuel uncovery ~~dewatering kinetics~~, hydrogen production, vessel failure, risk of explosion, risk of basemat penetration by corium, and the amplitude and kinetics of radioactive release.	Unclear, what is meant with fuel dewatering.		X...fuel uncovering kinetics…		It is not just the fuel uncovering, but also its evolution in time.
China	12	6,8	Note 17 suggest to delete	There is no severe accident due to core damage in Chinese high temperature gas-cooled reactor	X			
China	13	6,9		The severe accident codes used now cannot simulate reactivity accidents. If it is necessary, what should be done .It is suggested that IAEA can give cases or practices for reference			X	The safety guide does not intent to give cases or practices but recommendations.
Bulgaria	31	6,10	The analysis of the progression of severe accidents ~~inside~~ for the reactor should be performed using one or more computer codes for severe accident simulation (see Annex I for examples of computer codes for water cooled reactors).	The statement is valid for both, inside and outside of the reactor. Therefore, it is better not to specify for which phase of accident progression how computer codes should be used.	X			
Saudi Arabia	30	6,11	*(c) The code(s) should be verified and validated against the severe accident phenomena treated by them (validation matrix should be available).* *(d) The validation and benchmarking effort and the associated documentation should be sufficient to support the necessary severe accident analyses (see for example Ref. [16])*	The order of the recommendations is more logical as proposed.	X			
Finland	12	6,12		Should be removed. Repetition of paragraph 6.5.			X	There is no repetion, but complement. 6.5 provides recommendation on the appropriate selection of the analysts while 6.12 provides recommendations on the knowledge the analysts need to have to use the code.
ENISS	5	6,13	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 6.13 ...sufficient safety systems or safety features		In compliance with SSR-2/1 (Rev.1) Req 20
Finland	2	6,13	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "safety systems or safety features"		The term "safety systems" in this context refers to systems for all accident conditions, so the term is changed to "safety systems or safety features" in compliance with SSR-2/1 (Rev.1)
Saudi Arabia	31	6,13	The analyses of severe accidents should initially cover key sequences for each PDS leading either to a successful stable state of the plant, where sufficient safety systems *or safety features* have operated correctly so that all the required safety functions necessary to cope with the PDS have been fulfilled	Not all the involved systems are safety systems.		X ...successful ~~stable~~ controlled state of the plant, where sufficient safety systems or safety features have operated correctly so that all the required safety functions necessary to cope with the ~~PDS~~ sequences have been fulfilled, or to filtered containment venting (if provided) or to a degraded state (footnote 18 added) as Degraded state is here considered as a state following a severe accident where the safety functions performed by the containment and its associated systems are affected.		Term "controlled" state added in compliance with Requirement 20 of SSR-2/1 (Rev.1). Footnote 18 added to define "degraded state". The term "PDS" was changed to "sequences" for correct wording.
Bulgaria	32	6,14	The analysis of individual phenomena should be supported by severe accident analyses as needed ~~by the analysts performing those individual analyses~~ (see also Section 10)	The "by the analysts performing those individual analyses" seem to be excessive.	X			
Bulgaria	33	6,14	(b) Interaction of core, core structures and corium with coolant inside and outside the reactor pressure vessel (in-vessel and ex-vessel steam explosion);	Does this refer to steam explosion? If so, better to clarify it.		X...(b) Interaction of core, core structures and corium with coolant inside and outside the reactor pressure vessel (e.g. quenching and steam production) and induced effects on the plant;		Clarification provided.
China	14	6,14	Some examples of individual phenomena for water cooled reactors are provided below:(a) Structural-mechanical behavior of components of reactor coolant system in the event of high-pressure severe accident scenarios;(b) Interaction of core, core structures and corium with coolant inside and outside the reactor pressure vessel;(c) Ex-vessel cooling of the reactor pressure vessel for in-vessel melt retention;(d) High pressure melt ejection and direct containment heating;(d) Hydrogen and carbon monoxide generation, flow and distribution in the reactor containment and mitigation means to cope with combustion behavior; (e) Ex-vessel corium cool ability; (f) Criticality accidents effects;(g) Containment pressurization.	High pressure melt ejection and direct containment heating are suggested to add into the list.	X			
Bulgaria	34	6,15	...may not be useful or ~~productive~~ appropriate in severe accident analyses for Level 2 PSA because...	It seems better to use term "appropriate" instead of "productive".	X			

Saudi Arabia	32	6,15	In general, the analyses should be performed in a best-estimate manner regarding applied codes, models, model parameters, as well as boundary conditions. Conservative assumptions for the severe accident analyses, which are common use *for design basis accident* analysis ~~the design of nuclear power plants~~, may not be useful or productive in severe accident analyses for Level 2 PSA because, for example, conservative assumptions may distort the results and risk insights,	Severe accidents are currently considered in the design of new nuclear power plants.	X			
Bulgaria	35	6,16	~~The integral severe accident analyses can also be used for the determination of the source terms (see also Section 9).~~	The statement is excessive. This information is already provided in para. 6.2	X			
Germany	25	6,16	The integral severe accident analyses can also be used for the determination of the source terms (see also Section ~~9~~10).	Editorial. Reference should be "Section 10"			X	See comment 35 of Bulgaria
Pakistan	4	6,17	**Mitigation measures for s**evere accident management ~~measures for both prevention of core damage as well asmitigation~~ should be considered in the severe accident analyses with realistic timing for humanactions.	As para 6.8 to 6.18 are related to 'analysis of severe accidents involving reactor core damage'and as per IAEA safety glossary 2022, "Severe accident management" is limited to mitigation of the consequences of a severe accident.		X Accident management measures for both prevention of core damage as well as mitigation should be considered in the severe accident analyses with realistic timing for human actions.		The text is modified to consider both prevention and mitigation for accident management, considering that it is not just for severe accident.
Germany	26	6,18	All severe accident analyses (description of input decks, boundary conditions, assumptions, results) should be part of the Level 2 PSA documentation …	Extension. Also, the input deck and boundary conditions for each analysis must be described.		X...6.18. All severe accident analyses (e.g. description of input decks, boundary conditions, assumptions, results) …		The "e.g." is added since the list might not be exhaustive (e.g. description of main sources of uncertainties)
Germany	27	6,20	Depending on the plant configuration (e.g. for plants with the spent fuel pool located inside the reactor building, whether the spent fuel pool is inside or outside the reactor containment ~~building~~), severe accident analysis should provide information on the interactions between the reactor and the spent fuel pool: there may be mechanisms whereby a reactor accident can induce a spent fuel pool accident and vice versa.	For more clearness: For plants with the SFP located inside the reactor building it has further to be distinguished if the SFP is inside or outside the containment.		X… spent fuel pool location plant configuration (e.g. inside the reactor containment, outside the reactor containment but inside the reactor building, outside the reactor building		
Bulgaria	36	6,20	The outcome of this analysis may require ~~could be some~~ additional accident scenarios (involving both reactor and spent fuel pool) being added to the Level 2 PSA	The idea is clear, but the way is presented is a little bit unexpected.		X...The outcome of this analysis somemay be consideration of additional accident scenarios …		Modified for better reading.
ENISS	5	6,22	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 6.22 d ...use common systems, and…		In compliance with SSG-15 (items important to safety)
Finland	2	6,22	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "common systems"		The term "safety systems" in this context refers to systems for common to both thre actor and the spent fuel pool, so the term is changed to "common systems" in compliance with SSG-15 (items important to safety).
Germany	28	6,22	Interconnection between the reactor and the spent fuel pool might be possible (e.g. the possibility to use common safety systems, and common severe accident management ~~guidelines~~ strategies), …	Clarification	X			
Saudi Arabia	33	6,22	Interconnection between the reactor and the spent fuel pool might be possible (e.g. the possibility to use common ~~safety~~ systems, and common severe accident management guidelines strategies), decay heat loads in each of these locations, and possible fuel assemblies handling.	Not all common systems are safety systems.	X			
Bulgaria	37	6,24	The analyst(s) should identify any possible lack of information on plant design or procedures and any lack of information from systems and components qualification~~, including ageing effects~~.	Aging effects should be part of a dedicated analysis and should cover both L1 and L2 PSA. Therefore, mentioning here raises more questions than to provide guidance.	X			
Finland	13	6,26	A plant specific list of uncertain parameters to be varied in the frame of the uncertainty/sensitivity analysis should be derived. Extreme care should be used, if including parameters such as correlation coefficients, model parameters, etc. used in modelling the phenomenology of severe accidents in the corresponding computer codes, which are established as part of the computer code validation procedure. Otherwise, their variation can lead to completely incorrect results of the uncertainty analysis.	Many correlations have significant uncertainties. These uncertainties can and preferably should be included in the uncertainty/sensitivity analysis, but the range of possible variations needs to be very carefully determined.		X… Care should be used, if including The list of parameters for uncertainty analysis should not include parameters such as correlation coefficients, and model parameters, etc. used in modelling the phenomenology of severe accidents in the corresponding computer codes, which are established as part of the computer code validation procedure.		Better reading.
Germany	29	6,26	A plant specific list of uncertain parameters to be varied in the frame of the uncertainty~~/~~ and sensitivity analysis should be derived.	Editorial	X			
Russian Federation	3	6,26	A plant specific list of uncertain In order to avoid obtaining incorrect parameters to be varied in the frame of results of uncertainty analysis when the uncertainty/sensitivity analysis should varying parameters. be derived. At forming the list of parameters for uncertainty analysis, it should not include as parameters correlation coefficients, model parameters, etc. used in modeling phenomenology of severe accidents in the corresponding computer codes, established as part of the computer code validation procedure. The formation of a list of parameters to be analyzed for uncertainties should be based on accepted criteria for selecting parameters for uncertainty analysis. Otherwise, their variation can lead to completely incorrect results of the uncertainty analysis.	In order to avoid obtaining incorrect results of uncertainty analysis when the uncertainty/sensitivity analysis should varying parameters.		X...The establishment of a list of parameters to be analysed for uncertainty should be on the basis of accepted criteria for the selection of parameters for uncertainty analysis.		Better reading.
Germany	30	6,27	Table 4: Corium stratification inside vessel lower plenum (metallic/oxidized layers, focusing effect for thermal flux)	For more clearness	X			

Germany	31	6,27	Table 4: Heat-up and creep rupture of reactor coolant system pressure boundary (hot leg nozzle, pressurizer surge line and steam generator tubes). Impact of possible steam generator tubes ~~defaults~~ flaws.	Clarification	X			
Germany	32	6,27	Table 4: Hydrogen combustion simultaneously with heat transfer to containment atmosphere (pressurization). Releases of radioactive material	Editorial. Carriage return after the delimiter. “Releases of radioactive material” – new line	X			
Germany	33	6,27	Table 4: Generation of ~~incondensable~~ non-condensable and/or combustible gas …	Editorial.	X			
Germany	34	6,27	Table 4: Mixing and/or stratification of flammable gas in containment atmosphere Local increase of concentrations e.g due to strengthened steam condensation in cold containment regions. Steam or nitrogen inerting	Additional phenomena, please add.	X			
Japan	15	6,27	Table 4: EXAMPLES OF AREAS OF UNCERTAINTY RELEVANT TO THE PROGRESSION OF SEVERE ACCIDENTS FOR ~~INSIDE~~ WATER COOLED REACTORS	Editorial.	X			
Russian Federation	4	6,27	Table 4, Table 9 No	Often, information from IAEA documents, given as an example (for example, Table 4 in this guide), is considered by the user as a guide to action, the provisions of which must be followed exactly, which is not entirely correct. If these provisions are not precisely defined in the guide, then there is a collision in their practical application. In this regard, in order to specify the information, it would be appropriate in Table 4 and Table 9 for each of the phenomena listed in column 2 to provide an approximate list of parameters subject to uncertainty assessment (add a third column to table 4). For the same reasons, it is proposed to move table 4, as well as table 9, to a separate appendix to this guide. The suggestion might make it possible to avoid the problems indicated in the commentary on the practical application of the data from tables 4 and 9. The data in tables 4 and 9 themselves are examples that essentially detail the phenomenology of severe accidents. Complementing those tables with examples of specific parameters for each item from tables 4 and 9 does not impose restrictions related to the development of the documents having safety standards status.			X	Table 4 and table 9 are proposed as example of areas of uncertainties only. Adding a third column to tables 4 and 9 with the list of related parameters to the phenomena listed was neither forseen nor proposed in the process of development. If the list is provided, it could be considered for addition. Besides, other IAEA publications could provide that information.
Pakistan	5	7,2	The capability of the containment to maintain its leak tightness and **structural integrity** under internal loads (Para 7.4-7.11)	The structural integrity or strength assessment of containment against loading conditions to assure the confinement function of containment is not described as part of containment integrity analysis. Only leak tightness function is assured. Moreover, referred Para 7.7 deals with strength assessment of containment. Hence, both leak tightness and structural integrity should be described in Para 7.2.	X			
Canada	4	7,4	Include a sample containment failure fragility curve.	To provide some perspective into failure probability of containment as a function of pressure and temperature.		X…fragility (hyper)surface (see Ref. [80] for examples).		Reference to NUREG-6906 was added where examples are provided.
Pakistan	6	7,6	Table 5: Materials~~used~~**&Its Properties**	For structural design and analysis of containment, material properties (elastic or plastic range) are required as input for modelling. Therefore, the term material properties may be replaced by “**Material &Its Properties**” for clarity.	X			
Bulgaria	38	7,10	As an example, the behaviour of the ~~material access closure system~~ confinement structure materials should be studied under severe accident conditions	The terminology is not very clear. What is the meaning of "material access closure system"?		X...As an example, the behaviour of the equipment hatch closure system should be studied under severe accident conditions …		To be in accordance with terms in SSG-53.
Saudi Arabia	34	7,10	Large penetrations (e.g. ~~material access, personnel access~~ *equipment hatches, personnel hatches*) and singularities zones can be a relative weak point of the reactor building in severe accident conditions	For consistency with Table 5 and other paras, e.g. para. 7.21.	X			
Pakistan	7	7,10	Large penetrations (e.g., ~~material~~ ~~access~~**equipment hatch,** personnel **hatches** ~~**access**~~) …… As an example, the behaviour of the material accessequipment hatch closure system….	In table 5, the terms “equipment hatch” and “personnel hatch” are used. The same terms may be used here for consistency.	X			
Germany	35	7,13	…. This may be of particular concern for the response of the containment basemat but also, ~~according to~~ depending on the plant design, the response of the containment wall, or the reactor pressure vessel support structure (e.g. concrete pedestal).	Clarification	X			
Saudi Arabia	35	7,15	For example, the response of a reactor pressure vessel support structure (e.g. concrete pedestal), containment wall or floor to the complete or partial penetration by core debris *should be examined if calculations of severe accident progression suggest such levels of erosion are possible.*	The 2nd sentence of para. 7.15 is not complete.	X			
Germany	36	7,17	The potential for containment isolation failure should be assessed. All the containment penetrations should be modelled or a careful justification ~~has to~~ should be provided to justify ~~the screen out of~~ that some penetrations are screened out.	Please make “should” statement.	X			

Saudi Arabia	36	7,17	For instance, containment isolation may not be modelled for normally closed lines provided that isolation valves would not be opened during the accident (e.g. due to the initiating event or type-A human failure event *(see para. 8.1)*) or for closed loop systems inside the containment provided that closed loop integrity will not be threatened during the accident.	Type-A human failure event is defined in para. 8.1 and not before.	X			
Bulgaria	39	7,19	The potential for containment bypass (release from the core to the environment without being able to credit containment) should be assessed ~~if not already provided~~ through interface with the Level 1 PSA (see Section 5) or through APET logic (see Section 9).	The bypass sequences identification is one of the major tasks of interface. Additional bypass sequences should be identified in APET/CET logic structure.	X			
Finland	14	7,19	The bypass paths should be identified by a rigorous search of systems located outside the containment and linked to reactor coolant loops or containment atmosphere.	Some lines can be connected to containment atmosphere and can cause bypass. Examples are fuel pool cooling lines, HVAC and suction for ECCS or containment spray. These are potential bypass locations although they are not the cause for initiating event. These lines should be mentioned.	X			
Bulgaria	40	7,21	If this condition is present, the assessment of containment internal loading and the effects on concrete structures due to molten core debris interactions is not applicable to the assessment of containment integrity since the containment is in a bypassed state.	I do appreciate this statement. Nevertheless, this statement contradicts the statement in paras. 6.13 and 10.8 where are discussed more than one containment failure modes. The guidance should be clearer. For example, if the subsequent containment failure will not cause significant change in releases to the environment, then further assessment of these containment failures is not needed.			X	There is no contradiction since this para highlights the applicability of recommendations when the containment is opened.
Saudi Arabia	37	7,24	Statistical feedback based on test samples from construction sites may be useful to assess material variabilities. Benchmark from mock-ups or feedback experience from pressure tests (if available) may be useful to assess modelling uncertainty. *Modelling* uncertainty may be assessed via reference to Sandia large scale simulated containment failure experiments (see Ref. [19])	For consistency		X… One example for assessing these m~~M~~odelling uncertainty may be ~~assessed~~ via reference to Sandia large scale simulated containment failure experiments (see Ref. [19]).		For better reading.
Germany	37	7,25	… Each fragility curve should be characterized by a best estimate (median) failure pressure, a parameter representing the material variability and a parameter representing the modelling uncertainty (see para. 7.23 ~~0~~).	Placeholder '0' for paragraph number has to be replaced.	X			
Saudi Arabia	38	7,25	Each fragility curve should be characterized by a best estimate (median) failure pressure, a parameter representing the material variability and a parameter representing the modelling uncertainty ~~(see para 0)~~.	Para. 0 does not exist. Please consider either correct the number of the reference para. or remove it.		X...modelling uncertainty (see para 7.23~~0~~).		Para number updated.
Saudi Arabia	39	7,30	Severe accident phenomena modelling for induced containment bypass ~~(see Section 6)~~	The reference to Section 6 is not correct. Please consider using a correct reference or remove the reference to Section 6.		X...Severe accident phenomena modelling ~~for induced containment bypass~~ (see Section 6)		The reference to severe accident modelling is important.
China	15	8,2	c)strategies and guidelines for deployment of non-permanent equipment or additional strategies not considered in emergency operating procedures and SAMG, if such strategies and guidelines have been implemented.	For level 2 PSA, SAMG strategies have considered part of non-permanent equipment. The content here focus on the strategies that may not be included in EOPs or SAMGs, which can still be considered in Level 2 PSA modeling.	X			
Bulgaria	41	8,4	~~Depending on the objectives and intended uses with Level 1 PSA, it is advised to revise the PSA level 1 human reliability assessment to reassess level 1 operator actions from a Level 2 PSA perspective (e.g. conservatism may have been used resulting in too high numbers).~~	The idea is not very clear. It seems that all human actions need to be revised. Changing the HEP in L1 PSA will change the results in L1 PSA. Why this should be done here? Maybe this is feasible only in some specific cases? Anyway, this seem to be relevant for L1 PSA guidance for lowering the conservatism and no that related to L2 PSA.		X 8.4. Some operator actions considered in the Level 1 PSA human reliability analysis may be considered for applicability to the Level 2 PSA. Such human actions may be considered failed in the context of Level 1 PSA but could be become feasible in the Level 2 PSA considering an extended time window available. This is because the criteria for core damage considered in Level 1 PSA is more restrictive than the criteria applied in Level 2 PSA for arresting the accident progression.		This is related to those operator actions whivh have an impact in Level 2 PSA.
Saudi Arabia	40	8,4	Depending on the objectives and intended uses with Level 1 PSA, it is advised to revise ~~the PSA~~ *Level 1 PSA* human reliability ~~assessment~~ *analysis* to reassess *the corresponding* operator actions from a Level 2 PSA perspective (e.g. conservatism may have been used resulting in too high numbers)	More precise formulation.	X			
Saudi Arabia	41	8,7	How human actions are prescribed. Depending on the organization put in place to deal with a severe accident, some actions may be carried out independently by the plant staff, while the others need to be approved or directed by the ~~crisis organization~~ *technical support centre* . In the latter case, this requires the ~~crisis organization~~ *technical support centre* to be fully operational and a good coordination with plant staff;	Consistency with SSG-54	X			
ENISS	9	8,10	It is important to ensure that potential dependencies between operator actions should be assessed and taken into account when appropriate. This includes the dependencies between the human actions credited in Level 2 PSA and the dependencies between the human actions credited in Level 1 PSA and Level 2 PSA**,** ~~noting that strong dependency can occur if the human actions are performed by the same operators, if they involve the same equipment, or if the actions are close in time~~. Degree of dependency can be influenced in particular by the organization and procedures that are implemented on the NPP, the context of each human action (preventing core melt or mitigating severe accident), the delay between the human actions and if the human actions are performed by the same operators or if they involve the same equipment**.**	Introduction of a more general and neutral wording, avoiding any judgment ("strong dependency").	X			

Bulgaria	42	8,13	8.13-8.17: The mission time for all the SSC included in PDS bridge trees and/or APET should be justified. The basis for it is usually derived from severe accident progression analyses. The mission time for L2 PSA might be quite longer than the one used in L1 PSA. Therefore, the MT for SSC credited L1 PSA, which are expected to impact the accident progression (especially those that are expected to ensure stable end state) should be revisited and changed as needed.	It will be good to dedicate at least one paragraph on the mission time for the systems and the difference from the L1 PSA. Note that even for the systems that has already been included in L1 PSA (included after that in bridge trees), the MT also need to be reconsidered.		X 8.18. The mission time of each SSC credited in Level 2 PSA APET should be defined accordingly to role of the SSC during the severe accident progression until the plant reach a controlled state (see para 5.5). The basis for it is usually derived from the severe accident progression analyses. The SSC mission times for Level 2 PSA defined in this way may be different than or the same as those used in Level 1 PSA.		
Bulgaria	43	8,13	8.13-8.17: The success criteria for SSC included in L2 PSA should be justified based on the results from severe accident progression analyses. The success criteria for some systems are expected to be different compared to ones used in L1 PSA (e.g. 1/3 LPIS is generally accepted success criterion for LPIS, which is expected to be inappropriate in L2 PSA due to additional heat sources).	It will be good to dedicate at least one paragraph on the success criteria for the systems and the difference from the L1 PSA.			X	The success criteria is already covered in para 5.5
China	16	8,13	The description "Equipment reliability in a Level 2 PSA is usually modelled using the same techniques as applied in the Level 1 PSA" modifies to "Equipment reliability as well as common cause failure in a Level 2 PSA is usually modelled using the same techniques as applied in the Level 1 PSA".	Add the guidance about common cause failure.		X… 8.13. Equipment reliability, including common cause failures,		Text changed since common cause failures are included in equipment reliability data.
Pakistan	8	8,14	Assessment of the reliability of equipment credited within the Level 2 PSA ~~should~~**may not** consider the periodic testing and maintenance practices or planned procedures.	In level-I PSA we generally model unavailability due to periodic test and periodic maintenance. Whereas in level-II PSA, the plant is already is in accident condition as starting point of level-II PSA is after core damage. Therefore, failures of planned activities in level-II PSA modelling may be neglected due to substantial time.			X	System unavailability due to planned activities in those systems needed for Level 2 PSA needs to be considered since those systems might be different than those considered in Level 1 PSA. (e.g. containment associated systems)
Germany	38	8,15	… Adverse environmental impacts may include containment and/or auxiliary buildings high temperature, pressure, …	Editorial	X			
Saudi Arabia	42	8,15	Examples of adverse conditions that could affect equipment reliability are energetic events (e.g. short term temperature and pressure spikes or impulse loadings from detonations or steam explosions) or high radiation environment *that can impact specific SSCs* (e.g. the electronic instrumentation, rubber gaskets that could be vulnerable to high radiation).	The given examples are referring to SSCs and not to radiation environment.	X			
Hungary	9	8,16	Repair actions should be credited in Level 2 PSA only if there is strong justification for their feasibility. It might be possible to credit repair actions if the specific failure mode of the equipment is known for the specific sequence and (i) it is possible to diagnose the failure, (ii) the spare parts and repairing personnel are in place, (iii) the environmental and work conditions needed for performing repair are given or they can be ensured, and (iv) the time window is sufficiently long to credibly assume the possibility for repair, including the time needed to bring spare part and repairing personnel to the plant, (v) the premises where there repair will take place can be reached by the operators. Reliable data should moreover be used to assess credible probabilities of repairing components and systems. For the components that are not reparable after a severe accident occurrence and that are continuously required after core melt (for corium cooling, for example), their failure probability assessment should integrate this long mission time. A discretization of the failure modelling for different time windows could be implemented to consider different consequences as a function of the instant of failure.	I think point (iii) does not specify that you not only need the area where the repair will take place to be safe, but also the route the mechanics/electricians/etc. will have to go through in order to get to a specific room. For example if there is an ECCS pump room in the basement, which is safe at the moment, but the corridors leading to it are not, then it cannot be repaired because the operator personal cannot get there. Therefore I suggest to add this one more case (v) to the list.		X… (iii) the environmental and work conditions needed for access and performing the repair are given or they can be ensured,		(iii) is improved to catch the recommendation.
Saudi Arabia	43	8,19	[…], operator training, and the coordination between the plant staff organization and ~~crisis organization~~ *technical support centre staff* after entering severe accident management guidelines	Consistent with SSG-54 terminology.	X			
China	17	8,21	As for the uncertainty of equipment reliability in the case of severe accidents, cases or good practices of in various countries is suggested to given.	How to deal with the uncertainty of equipment reliability in the case of severe accidents is indeed a problem, which is very difficult since there is no special reliability database for the equipment under severe condition.			X	The safety guide provides recommendations on what should be done, in this case related to uncertainties about equipment reliability for severe accident conditions, but it can't provide information such databases on this topic.
France	8	9,1	In para 2.6.	Typo correction	X			
USA	3	9,4	Suggest rewording Note 22 as follows: "The ASME/ANS Level 1 PSA Standard [23] describes all the technical elements necessary for developing event trees capable of assessing large early release frequency. In the United States the large early release frequency metric is used in regulatory risk-informed decision-making."	While the PRA standard is worded as requirements, in the US the PRA standard is considered guidance, and so it would be preferable to avoid the word "requirements" to avoid implying regulatory requirements. Secondly, added clarification that the large early release frequency is used in risk-informed decisions.	X			

Bulgaria	44	9,6	No description is provided for Phase 1 and 2 is provided.	The information is provided only for Phases 3 and 4. For phase 1 and 2 no discussion is provided. This makes the text imbalanced.		X...9.6. 9.6. Phase 1 is the initial period of in-vessel core damage (i.e. fuel rod heating up above criterion for temperature related to fuel integrity, generalized oxidation process of fuel cladding, and start up control rods melting). Phase 2 typically starts around the time of core melting and relocation to the lower plenum of the reactor pressure vessel (i.e. formation of debris from melting of fuel cladding, reactor pressure vessel internals and fuel, relocation of debris and melt in the lower plenum of the reactor pressure vessel). Phase 3 starts at is close to the time of reactor pressure vessel failure and covers the phenomena that occurs immediately after (to address challenges occurring due to failure of the reactor pressure vessel, e.g. direct containment heating, ex-vessel steam explosion).		Text added describing Phases 1 and 2.
Germany	39	9,6	… Phase 4b is the long term, starting from a few hours after failure of the reactor pressure vessel (to address challenges arising from ex-vessel melt behaviour, …) (see paras 7.12-7.22).	Insertion of a missing paragraph number.	X			
Saudi Arabia	44	9,6	[….], e.g. pressurization due to the generation of non-condensable gases during core–concrete interaction or combustion phenomena or pressurization due to ongoing steam generation, human actions and equipment behaviour) (see ~~paras 7-7.22~~ para. 7.22).	There is only one paragraph 7.22.		X...equipment behaviour) (see paras 7.1~~7~~-7.22 and Section 8). Typical		To provide link to appropriate topics in the SG.
Saudi Arabia	45	9,6	Table 6 - item 24: Do sprays actuate or *are they* restored to operate in the short time frame?	Editorial: better formulation.	X			
Saudi Arabia	46	9,6	Table 6 – item 25 Do fan coolers actuate or *are they* restored to operate in the short time frame?	Editorial: better formulation.	X			
Germany	40	9,8	The rationale used to develop appropriate probabilities for each branch can sometimes be made more traceable by decomposing the problem into a number of sub-issues ~~according to~~ in accordance with the governing phenomena (see Refs [32], [33]).	Editorial	X			
Canada	6	9,10	(a) Suggest providing some examples of "basic principles".	It is not clear what basic principles means in this case.		X… (a) Basic physical principles calculations (e. cooling water flow rate compared to core decay heat)		Example proposed
China	18	9,11	The description "Thermally induced steam generator tube rupture:" modifies to "Thermally induced and pressure induced steam generator tube rupture".	In severe accident, there two type induced SGTR, i.e. thermally-induced and pressure-induced SGTR.			X	Pressure induced STGR are relevant for Level 1 PSA
China	19	9,11	Delete Table 6 and replace with the following figure that is derived from "Fig. 6 Generic event tree with functions and containment failure modes" in ASAMPSA2 BEST-PRACTICES GUIDELINES FOR LEVEL 2 PSA DEVELOPMENT AND APPLICATIONS, volume 2	Based on the mention "the state of knowledge of severe accident phenomena has progressed since the NUREG-1150…thus reducing its usefulness as a reference for modern Level 2 PSA studies…A compilation of recent, relevant severe accident phenomena can be found in Refs [36], [37], as well as work associated with (ASAMPSA2) project" in 9.11, however, the example in table 6 is based on NUREG-1150, so the example should be modified based on ASAMPSA2.			X	Para 9.11 makes reference of updated publications related to different phenomenon relevant for Level 2 PSA, including the ASAMPSA 2 Project. In addition, figure 6, of that study, mentioned in the comment does not cover all the nodal questions for all phases, as currently presented in Table 6 which has been particularly revised and updated based on the updated information from best recognized practices.
Germany	41	9,11	… NUREG-1150, ~~Reference~~ Ref. [22], has historically been a key source of information for many Level 2 PSAs. However, the state of knowledge of severe accident phenomena has progressed since ~~the NUREG-1150, Ref. [22]~~ that study, thus reducing its usefulness as a reference for modern Level 2 PSA studies, …	Editorial	X			
China	20	9,12		This paragraph describes the containment fragility models, it's suggested to add into Section 7.			X	Para deleted since it is already covered in para 7.4.
Saudi Arabia	47	9,12	The number of new chemicals and substances in the plant should be minimized. However, ~~he~~ *the* replacement of harmful chemicals or other substances […]	Editorial			X	Not clear, the text does not correspond to 9.12.
Saudi Arabia	48	9,12	Experimental programmes regarding the response of containments to internal pressurization conditions beyond *their* design basis that may be useful […]	More precise formulation	X			
Saudi Arabia	49	9,13	The assignment of numerical values is thus indicative of the ~~analyst's~~ analysts' confidence in the rigour, applicability and completeness of deterministic predictions of relevant phenomena.	For Level 2 PSA, the analysts are usually more than one, due to the complexity of the analysis. We should avoid giving the impression that one analyst is enough.	X			
Saudi Arabia	50	10,1	On the other end of the spectrum *of objectives*, only the frequency of accidents that would result in a large early release may need to be characterized	More precise formulation.	X			
Bulgaria	45	10,2	10.2-10.3: Change the order of paragraphs	The way the paragraphs appear seem to be illogical. It seems more reasonable to start with APET endstates and then to explain about ST and RC tasks.	X			
Germany	42	10,2	… These release categories are each identified by a set of characteristics that impact the amount of radiological release that will arise for accident progression event tree sequences ~~matching these characteristics~~ (see paras 10.5 – 10.6)	For more clearness: Sentence is hard to understand. The last part of the sentence "matching these characteristics" seems unnecessary and makes the sentence complicated to understand.	X			

Germany	43	10,2	…. In the second sub-task, for the calculation of radiological release for each release category, code calculations are performed …	Editorial	X			
Germany	44	10,3	… Preliminary list of representative severe accident scenarios should be based on severe accident scenarios established for identified PDSs (see para 6.8)	Clarification: please use "scenarios" in plural	X			
ENISS	5	10,5	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 10.5 e ...of safety credited systems…		In compliance with SSG-3
Finland	2	10,5	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "credited systems"		The term "safety systems" in this context refers to system considered in PSA, so the term is changed to "credited systems" in compliance with SSG-3
Saudi Arabia	51	10,5	The availability of ~~safety~~ systems able to reduce radioactive releases (e.g. containment spray system, filtered containment venting system, suppression pool, ice condensers);	Not all those systems are safety systems.		X...(e)The availability of ~~safety~~ credited systems		Credited systems terminology is more appropriate, meaning those systems that are considererd for Level 2 PSA.
Bulgaria	46	10,6	TABLE 7 Time elapsing since the start of the severe accident* Add the following comment below the table: Comment: *Time should be related to the site-specific emergency (evacuation) plan	This time should be related to the site-specific emergency plan. Thus, the hours that are given in the next column, should be justified against real hours for the specific site from drills. Moreover, this should be part of typical L2 attributes. Otherwise, will be difficult to define LERF and LRF.	X			
China	21	10,6	TABLE 7 "Pressure of reactor pressure vessel during core damage" release attribute is suggested to be taken place by the "Pressure of reactor pressure when vessel failure"	Although "Pressure of reactor pressure vessel during core damage" has an impact on the source term, it is not the key impact and many plant designed with special pressure relief measures for severe accident. According to practice, "Pressure of reactor pressure when vessel failure" has a more critical impact on the source term.			X	According to the parctice it is not related only to the time when the reactor pressure breaks.
Finland	15	10,6	TABLE 7: Secondary containments Reactor buildings Suppression pools Overlying water pools Passive containment coolers Ice beds 'Tortuous' release pathways Submerged release pathway Alkaline materials	Passive containment coolers could be mentioned under "Passive engineered features providing capture mechanisms for radioactive material". Cooler main function is condensation to decrease containment pressure, but they have also some impact to fission product removal from containment atmosphere into water pool.	X			
Hungary	11	10,6	TABLE 7. Design basis conditions leakage Beyond design basis conditions leakage Catastrophic rupture of containment Loss of coolant accident in interfacing system Steam generator tube/tubes or header rupture Open containment isolation valves Open material hatch access Containment by-pass Basemat penetration	I think containment by-pass as a general phenomenon should be included in the list, even though the list contains certain sub-cases of containment by-pass.		X…Containment bypass: Loss of coolant accident in interfacing system Steam generator tube/tubes or header rupture		For better reading
UK	3	10,8	Additional clarification regarding containment failure modes: "Some accident scenarios can include several containment failure modes. The analyst should pay attention to the quantification of the frequency of each containment failure individually in order to capture their importance on the global results. **For slow containment overpressurization sequences the analysis should distinguish between containment leak and containment break as a leak can, with significant probability, prevent a major containment failure and subsequently limit the amount of the released fission products**."	When analysing the containment failure modes it is useful to add some explicit guidance to distinguish between containment leak and containment break in case of slow containment overpressurisation. In these scenarios there is a significant probability of containment leakage to develop and prevent large containment failure, hence leading to smaller amount of released materials.		X… New para as 10.9. For slow containment overpressurization sequences the analysis should distinguish between containment leak and containment break as a leak may prevent a major containment failure and subsequently limit the amount of the released fission products (see para 7.8).		Better to have it in a new para. The statement on significant is removed and reference to para 7.8 related to break preclussion and leak before break is provided.
UK	4	10,8	Suggest additional paragraph after 10.8: "The grouping of the accident progression sequences in release categories often requires some assumptions and simplifications to be applied which may introduce additional uncertainties. Special care should be taken to keep track of any assumptions and simplifications so that the additional uncertainties are not overlooked during the uncertainty analysis."	Similar to the first comment, when grouping the release categories, there will likely be a need to apply some assumptions and simplifications that increase the overall uncertainty of the results. The guide should provide some explicit directions to the analyst to keep track of the assumptions and simplifications and to address them in the uncertainty analysis.	X			

Japan	10	10,9	10.9. In Level 2 PSA, the source term specifies, for a given accident scenario, the amount and composition of radioactive material released from the plant to the environment and the timing, location and kinetic energy of the release. … 10.17. The analysis should be carried out for a representative accident sequence in each release category. Sensitivity analyses should be performed to provide confidence that the source terms have been accurately characterized and there is not an undue variation of the source term magnitude within each release category.	The distinction between "accident scenario" and "accident sequence" is unclear, and it seems there is no difference. If they are clearly used differently, clarify the definition.	X			Reference to para 11.22. The distinction between "accident sequence" and "accident scenario" in the different paras has been revised and corrected as appropriate.
Japan	11	10,9	These include fixed plant design characteristics, such as configuration of the fuel and the control assembly and material composition, core power density and distribution, fuel burnup and concrete composition as well as radioactive decay of radioactive releases. ~~These plant design characteristics will be the same for all the end states of the accident progression event tree.~~ The analyst should be familiar with the specific plant design features (see Section 4) and accident phenomena (see Section 6) for the definition of end states of the accident progression event tree.	The subject sentence is not understandable and not necessary.	X			
Saudi Arabia	52	10,11	[…] and intended applications of *Level 2* PSA.	More precise formulation.	X			
Germany	45	10,11	The extent to which source term analysis needs to be carried out depends on the objectives and intended applications of the PSA. If the source term is to be used within Level 3 PSA, the characterization of the source term should be sufficiently detailed to be adequate as an input for Level 3 PSA consequences calculations (see e.g. ~~see~~ Refs [48], ~~[46],~~ [49]).	Do we need IAEA Safety Glossary as reference here? We suggest to delete.		X… (e.g. see Refs. [48], [49])…		It is better to write "e.g." before.
Bulgaria	47	10,18	~~Work has been carried out on calculating the releases within a dynamic PSA framework (see Ref. [49]) where the releases and the accident progression area calculated together in an integral manner~~	Since the idea of the paragraph is not very clear and no connection is established with the rest of the paragraphs, it is recommended to delete the entire paragraph.	X			
Canada	7	10,18	"Work has been carried out on calculating the releases within a dynamic PSA framework (see Ref. [49]) where the releases and the accident progression area calculated together in an integral manner."	This is not a guidance or recommendation. The use of dynamic PSA mentioned is not an industry practice with current technology. Suggest rewriting this Para showing that it is a practice.	X			Paragraph deleted.
Germany	46	10,19	Source term calculations with integral computer codes for severe accident analysis, generally consider group categories of radioactive elements or chemical compounds rather than ~~on~~ individual radioisotopes (see Refs. [53], [54]).	Editorial	X			
ENISS	5	10,26	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.			X	10.26 b Here its correct to mention reactor safety systems or containment systems...
Finland	2	10,26	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.			X	10.26 b-remains as "...Differences in the operation of reactor safety systems or containment systems can invalidate…" since it is related to reactor safety systems and not to safety features.
Germany	47	10,29	The users of the computer code for source term analysis should be trained in the use of the code and be familiar with the phenomena being modelled by the code and the way that they interact, the meaning of the input and output data, and the limitations of the code. ~~Other r~~Recommendations on the selection of software, approaches and methods are provided in paras 3.15 ~~to~~ - 3.17.	Please use more specific wording, in line with the heading of the relevant subsection referred to here.	X			
Saudi Arabia	53	10,31	The source terms and frequencies of the release categories, the ~~later~~ *latter* obtained as a result of accident progression event tree quantification	Editorial.	X			
China	22	10,33	10.33,10.34	It is necessary to discuss the treatment of uncertainty calculation and sensitivity calculation of source terms and the selection of analysis parameters. It is difficult to analyze the uncertainty of source term, so it is suggested to give a practical cases.			X	Unfortunately, the safety guide can't provide practical examples for every single topic. Other documents could be developed to fill this gap.
ENISS	5	10,34	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 10.34...containment associated systems ...		In complicance with SSG-53
Finland	2	10,34	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "associated systems"		The term "safety systems" in this context refers to containment and associated systems in accordance with the SSG-53.
Saudi Arabia	54	10,34	[...], containment surfaces, and from scrubbing by containment ~~safety~~ systems are still subject to research	Not all those systems are safety systems.		X...containment ~~safety~~ associated systems are still …		To be in compliance with terminology used in Requirement 20 of SSR-2/1 (Rev.).
Saudi Arabia	55	10,35	Uncertainties associated with containment response to *conditions* beyond *its* design basis ~~conditions~~ lead to [….]	More precise formulation.	X			
Canada	5	10,36	Table 9: "Radioactive release into the environment with regard to containment break size, containment leak rate, released fraction of inventory, iodine chemistry, deposition of radioactive material onto containment surfaces."	Radioactive releases need to account for deposition of radioactive materials onto the containment surfaces for a realistic estimate of releases.	X			

Finland	16	10,36	Table 9: …- Deposition on surfaces and into pools inside the containment…	A new bullet should be added. Deposition on surfaces and into pools inside the containment may have significant effect on the amount of radioactive materials released to the environment especially in longer time frames.	X			
Finland	17	10,36	Table 9: Chemical processes in primary circuit, corium and during molten core–concrete interaction;	Fission products may have chemical interactions with each other in high temperature in primary circuit. During in-vessel melt retention there are chemical processes that may have impact to release from the melt.		X...•Chemical processes in reactor coolant system, corium and …		To be in accordance with the terminology og SSG-56
Russian Federation	5	10,36	Table 10: Fraction of initial core inventory to environment	Clarification. It is proposed to clarify the heading from Table IO "Fraction of core inventory to environment" and use it in the form of "Fraction of initial core inventory to environment".		X …Note added, It says: Fraction of core inventory to the environment here refers to core inventory before severe accident scenario begins.		Better reading, since "initial", might induce to the mistake to look at the core when it was loaded. Note added.
Russian Federation	6	10,36	Table 10: Design Leakage	Clarification. It is proposed to replace in Table 10 and other places of the document (if any) "Nominal leakage" with " Design Leakage ". It is proposed to at least explain here or where the term "Nominal leakage" first occurs, what is meant by this.		X…Note added as: The value of the "nominal leakage" refers to the normal operating conditions (measured by tests), which might be different than design		The value of the leakage refers to the normal operating conditions (measured by tests), which might be different (higher) than design Note added.
Russian Federation	7	11	Section 11 IMPORTANCE, UNCERTAINTY AND SENSITIVITY ANALYSES No	Add Separate Section. In order to systematize the information and avoid its repetition, all recommendations concerning the analysis of uncertainties from Sections 5 to 11 (pp. 5.13, 6.24-6.27, 7.23-7.30, 8.17-8.22, 10.34-10.36, 11.18-11.27) are proposed to be include in a separate Section "IMPORTANCE, UNCERTAINTY AND SENSITIVITY ANALYSIS". It is proposed to provide links to the new Section in Sections from 5 to 11. It is proposed to provide subheadings, corresponding to Sections from 5 to 11 and to indicate, how this information is associated with importance, uncertainty and sensitivity analysis. It is proposed to indicate whether these sources of uncertainty are identified correctly, how to quantify them. AII these tasks are a oart of the uncertainty analysis. In the current version of the guide and in the actual version of SSG-4, information related to uncertainty analysis is presented in an unsystematized form, which makes it difficult to use this information when it is scattered across different sections and paragraphs. In order to eliminate these shortcomings, it is proposed to implement the suggested above.			X	The structure of the safety guide in section 5 to 10 is to characterize the sources of uncertaninty and in an specific of section 11 the recommendations related to importance analysis, sensitiivy ananlysis and uncertainty analysis are already provided.
Saudi Arabia	56	11,3	As with the Level 1 ~~analysis~~ *PSA* , before quantifying the Level 2 PSA, care	More precise formulation.	X			
Saudi Arabia	57	11,7	The analysts should check that the accident sequences or cutsets identified by the solution of the Level 1 PSA model are propagated into the Level 2 *PSA* structure and are appropriately	More precise formulation.	X			
Canada	8	11,10	In support of the convergence study as noted in the last two sentences of this paragraph, please add the following footnote: "Convergence can be considered sufficient when successive reductions in truncation value of one decade result in decreasing changes in CDF or LERF, and the final change is less than 5%."	To provide a measure of convergence in support of appropriate selection of the truncation/cut-off value.		X...Performance of a study is a typical way to demonstrate convergence.		Text modified to be more clear.
Saudi Arabia	58	11,12	Title of Table 11 ~~MITIGATION~~ CONTAINMENT PERFORMANCE MATRIX ('C MATRIX')	The name of the Matrix is not consistent with para. 11.12.	X			
Germany	48	11,13	… This may not be sufficient and additional useful information may be presented such as the release categories frequencies for each plant operati~~ng~~onal state, the distribution of the different causes of containment failure for specific release categories.	Please change to "operational state", to be in line with IAEA Glossary	X			
ENISS	10	11,15	~~It is useful to summarize, T~~The contribution of each release category (R(n)) to the ~~large~~ total release ~~(early large release)~~ frequency, R, should also be tabulated, to enable identification of major contributors to the total release frequency.	Text improvement (as current sentence is confusing; moreover, according to the table 11, R seems to be a total release frequency).	X			
Saudi Arabia	59	11,15	As discussed above, by combining the results of the Level 1 PSA (frequencies of occurrence of the various PDSs and their associated uncertainties) with the conditional probabilities of various containment failure modes and/or release modes and their associated uncertainties resulting from quantification of the accident progression event tree;, the frequencies and uncertainties associated with each release category can be determined	There should be a coma after 'event tree'; otherwise, the first sentence would be incomplete.	X			
Saudi Arabia	60	11,15	It is useful to summarize, the contribution of each release category (R(n)) to the large release (early large release) frequency, R, *which* should also be tabulated, to enable identification of major contributors to the total release frequency.	More precise formulation.	X			
France	7	11,16	section 10	Typo correction	X			
ENISS	11	11,16	For each of the selected release categories, or related group of release categories, one representative accident sequence is selected for which a source term is estimated on the basis of results obtained from plant specific calculations employing an appropriate computer code for estimating source terms for severe accidents, as discussed in Section ~~0~~ 10 and Annex I, or past analyses from Level 2 PSAs of representative plants.	Section 0 does not exist. Reference to Section 10 seems appropriate.	X			

Germany	49	11,16	For each of the selected release categories, or related group of release categories, one representative accident sequence is selected for which a source term is estimated on the basis of results obtained from plant specific calculations employing an appropriate computer code for estimating source terms for severe accidents, as discussed in Section ~~0~~ 10 and Annex I, or past analyses from Level 2 PSAs of representative plants.	Placeholder '0' for section number has to be replaced.	X			
Saudi Arabia	61	11,16	[…], one representative accident sequence is selected for which a source term is estimated on the basis of results obtained from plant specific calculations employing an appropriate computer code for estimating source terms for severe accidents, as discussed in Section 10 and Annex I, […]	Section 0 does not exist.	X			
Saudi Arabia	62	11,16	When using representative plant analyses for releases, care should be taken to account for plant differences in core fission product inventory (typically associated with fuel design, core power and operational history, and containment failure modes and failure pressures). Considerations regarding the acceptability of source terms from representative plant specific PSAs should be documented.	If 'representative plant' has the same meaning as 'reference plant', there is no need to repeat the same recommendation as in para. 10.26.			X	It refers to analyses and not to the plant.
Germany	50	11,18	… These metrics may be more specific or may encompass more than one operating mode or operating state. Importance measures typically include: (a) the Fussell-Vesely importance ~~(F-V)~~; (b) the risk reduction worth; (c) the risk achievement worth and (d) the Birnbaum Importance metric.	The abbreviation 'F-V' is not further used in the Safety Guide, please delete.	X			
China	23	11,18	Suggest to delete "safety" or replace as" systems of concern" in the "Importance measures for basic events, groups of basic events, safety systems, groups of initiating events, etc"	Safety systems and other non-safety systems both modeled in PSA model. So, the importance measure suggests not limited to safety systems.		X...groups of basic events, credited systems, groups of initiating events		Credited systems terminology is more appropriate, meaning those systems that are considererd for Level 2 PSA.
ENISS	5	11,18	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given for example in SSR2/1.		X… 11.18 ..., credited systems...		In compliance with SSG-3
Finland	2	11,18	Replace *safety systems* with *items important for safety*	Safety systems do not include systems for design extension conditions. Correct term in many cases would be items important for safety, that cover safety related items, safety systems and safety features (for design extension conditions). Definitions are given in SSR2/1.		X…The term "safety systems" is modified as: "credited systems"		The term "safety systems" in this context refers to system considered in PSA, so the term is changed to "credited systems" in compliance with SSG-3
Saudi Arabia	63	11,18	Importance measures for basic events, groups of basic events, ~~safety~~ *used* systems, groups of initiating events, etc., should be calculated and used to interpret the results of the Level 2 PSA.	Not all involved systems are safety systems.		X… events, ~~safety~~ credited systems, …		Credited systems terminology is more appropriate, meaning those systems that are considererd for Level 2 PSA.
Finland	18	11,20	Delete 11.20	Use of PSA for improvements are discussed in paragraph 15. It is unnecessary to repeat the issue in many places.	X			
Bulgaria	48	11,21	Paragraphs 11.22 through 11.26 provide recommendations on meeting Requirement 17 of GSR Part 4 (Rev.1) [2] on uncertainty and sensitivity analysis for Level 2 PSA (issues giving rise to uncertainties are presented in TABLE 4 and TABLE 9). The uncertainties related to systems and operator actions are presented in SSG-3 (rev.1).	The paragraph should be completed with uncertainties related to system and operator actions uncertainties.	X			
Finland	19	11,22		It is not clear, what is meant by "initial fuel release". Please clarify.		X… Term modified to be "initial fuel damage"		In this para the term "initial fuel damage" refers to the initial radioactive material released from the fuel during the severe accident.
Saudi Arabia	64	11,22	In Level 2 *PSA,* ~~analyses~~ these grouped sequences can vary in the timing of the initial fuel release, impact of the event progression on the containment,	More precise formulation.	X			
USA	4	11,22	Delete sentence "so the Level 2 PSA should have extensive peer review".	This para 11.22 is in Section 11 dedicated to quantification and uncertainties, therefore requirements for peer review do not appear to be appropriate in this para 11.22 on incompleteness uncertainty. Peer review appears to be adequately captured in para 3.24		X…part of the sentence deleted, but text added as: errors~~, so the Level 2 PSA should have extensive peer review~~. Sensitivity analyses, including bounding analyses, may be employed to provide estimates regarding the significance of the uncertainty, so the Level 2 PSA should ensure that those sensitivity analyses are performed and reviewed (see paras 3.22 to 3.28).		It is important to recommend the need for peer reviews, so reference to relevant paras is added.
Saudi Arabia	65	11,24	The Level 2 PSA analysts should identify the dominant sources of uncertainty in the analysis and should quantitatively characterize the effects of these uncertainties on the baseline (point estimate) results.	This recommendation is not consistent with para. 7.67 of SSG-2 (Rev.1), which privileges sensitivity studies "since explicit quantification of uncertainties may be impractical due to the complexity of the phenomena and insufficient experimental data". Indeed, the sequences of DEC with core melting mentioned in SSG-2 (Rev.1) are part of the sequences of Level 2 PSA. If an explicit quantification of the uncertainties cannot be done for those DEC sequences, it cannot be done for the whole Level 2 PSA.Therefore, the recommendations related to uncertainty quantification/characterization need to be made consistent with para. 7.67 of SSG-2 (Rev.1).			X	There is no contradiction. The para does recommend to characterize uncertainties and that could be performed by either sentivitity or uncertainty analysis as mentioned in the last sentence.
Germany	51	11,26	~~Parameter/event/phenomenon specific~~ ~~s~~Sensitivity analysis, specific to parameters, events and/or phenomena, may be used to supplement a more comprehensive uncertainty analysis.	Clarification	X			

Russian Federation	8	11,26	Parameter/event/phenomenon specific sensitivity analysis may be used to supplement a more comprehensive uncertainty analysis. Sensitivity analysis is a useful tool to guide the selection of dominant sources of uncertainty. List of parameters/events/phenomena subject to sensitivity analysis should be based on the established list of plant specific list of uncertain parameters (see item 6.27). Example areas of uncertainty related to the progression of severe accidents are listed in table 4.	Clarification.			X	The purpose of the para is to provide recommendation for sensitivity analyses in general and not only for severe accident analyses. Table 4 already provides reference to those key parameters.
Canada	9	12,10	Add: "(e) Significant accident scenarios leading to large (early) release"	Significant accident scenarios are important to be included in the Level 2 PSA report.	X			
Bulgaria	49	12,15	~~The summary report should be prepared by an individual who has an excellent overview of the entire PSA study. It should be independently reviewed by individual task leaders and/or analysts for correctness and consistency.~~	It is recommended to remove the sentences, since their value to the entire guidance is doubtful. Moreover, the development of summary report could be a matter of teamwork and not just a person.		X...The summary report should provide a comprehensive overview of the entire Level 2 PSA study...		Text modified to be more clear in the recommendation.
Saudi Arabia	66	12,15	The summary report should be prepared by an individual who has an excellent overview of the entire *Level 2* PSA study	The recommendation is related to Level 2 PSA.	X			
Saudi Arabia	67	12,16	The relation between various parts of the *Level 2* PSA should also be included in this subsection of the summary report.	The recommendation is related to Level 2 PSA.	X			
Bulgaria	50	12,22	12.22-12.25 Entire paragraphs Examples for PSA maintenance recommendations: 12.22 The PSA should be maintained in a way to reflect the as-built and as-operated plant to support the applications for which it is being used or foreseen. 12.23 Changes in plant design or in operation procedures should be evaluated to de-termine whether such changes impact PSA model and documentation. Changes that would impact risk informed decisions should be incorporated as soon as practical.	Requirement 24 of the GSR Part 4 deals in more detail with the maintenance of safety assessment rather than communication. Moreover, in para. 12.14 it is already stated that summary report should be prepared for a wide audience, which means that all information provided there should not be proprietary. Therefore, it is more feasible to have some guidance how to maintain L2 PSA model and documentation and not how to communicate the results of it. It is recommended to remove current paragraphs.		X...on the communication of Level 2 PSA results to meet the Requirement 24 of the GSR Part 4 (Rev. 1) [2] related to the maintenance of the safety assessment (see para 5.9 of GSR Part 4 (Rev. 1) [2]). …		Those paras provide recommendations related to the communication to comply with para 5.9 of Requirement 24 of GSR Part 4 (Rev. 1) requiring: "*Consideration shall also be given to ways in which results and insights from the safety assessment may best be communicated to a wide range of interested parties, including the designers, the operating organization, the regulatory body and other professionals. Communication of the results from the safety assessment to interested parties shall be commensurate with the possible radiation risks arising from the facility or activity and the complexity of the models and tools used.*"
Saudi Arabia	68	12,22	*Paragraphs* 12.22 to ~~12.23~~ *12.24* present the recommendations to meet the Requirement 24 of the GSR Part 4 (Rev. 1) [2] related to the maintenance of the safety assessment.	Paragraph 12.24 is also relevant for the maintenance of the safety assessment.	X			
Saudi Arabia	69	13,3	In principle, the Level 2 PSA for the spent fuel pool is based on the same methodology as Level 2 PSA for the reactor ~~core~~ outlined in Sections 5-11. A spent fuel pool PSA	"core' is not necessary		X...pool Level 2 PSA study can be performed separate or combined with Level 2 PSA for the reactor ~~core~~, depending on		Better formulation
Saudi Arabia	70	13,4	A spent fuel pool PSA study can be performed separate or combined with PSA for the reactor *core* , depending on the specific needs and applications for developing the Level 2 PSA.	"core' is not necessary		X...pool Level 2 PSA study can be performed separate or combined with Level 2 PSA for the reactor ~~core~~, depending on		
Germany	52	13,5	…. Factors specific to spent fuel pool analysis include items which influence the accident progression and source term, such as: time since last core unloading ~~offload~~, the fuel loading in the pool (e.g. number of fuel assemblies, fuel burnup, fuel loading pattern), pool configuration …	Please put in line with other IAEA Safety Guides, where 'core unloading' instead of 'core offload' is used, see e.g. Safety Guides SSG-52, SSG-63, SSG-73 and SSG-82.	X			
Germany	53	13,6	The undesired end states (e.g. uncovering of fuel stored in the spent fuel pool or during fuel handling, boiling of the pool water) defined in Level 1 PSA for the spent fuel pool, as described in paras 10.2-10.6 of SSG-3 (Rev.1) ~~(paras 10.2-10.6)~~ [4], should be addressed in Level 2 PSA.	Editorial	X			
Germany	54	13,7	… Reactor accident sequences can impact the spent fuel pool, for example containment venting could accelerate boiling of the water in the ~~SFP~~ spent fuel pool if the ~~SFP~~ spent fuel pool is located inside the containment.	If you decide to use abbreviation SFP in this Safety Guide, please insert it in para 1.19, as this is the first appearance of the full term in the text. Otherwise please replace the abbreviation 'SFP' by the full term 'spent fuel pool' in the text.	X			SFP is introduced in para 1.19 based on comment 6 Germany. The acronym has been updated in the whole document.
Bulgaria	51	13,8	13.8-13.18: The entire section should be expanded with the guidance of the SFP specific modelling.	The section should be complemented with the following type of information: - What to consider when simulate both reactor and SFP? - What to consider when adopt arrangement of FA, esp. when you have several separated pools in SFP? What to consider when develop PDS? - What simplifications are applicable, what not? -What to consider in case of significant differences in time of core, fuel damage, - etc.			X	The interaction between the reactor and the spent fuel pool are provided in different paras from the reactor perspective (6.19 - 6.22). In addition, the status of practice does not allow to provide further recommendations particularly for the development of Level 2 PSA for the SFP, however references to reports are added for further information in para 13.1.
Bulgaria	52	13,8	13.8-13.18: Structure of the paragraphs	The structure does not cover all the relevant topics in SFP L2 PSA development. It is recommended to follow the structure adopted for MUPSA.			X	The structure follows the same structure proposed in this safety guide.
Canada	10	13,8	"Severe accident phenomena to consider in this analysis include~~s~~ heat transfer within the pool, fuel racks, ~~and to~~ surrounding walls~~,~~ and to the ground for SFPs that are below the ground level …"	For a realistic estimate of heat losses, heat transfer to the ground surrounding the walls should also be considered.	X			

Japan	12	13,8	To support Level 2 PSA development (if such a development is needed, see para. 13.2), deterministic analyses should be performed to analyse the severe accident progression in the spent fuel pool using one or more computer codes capable of modelling the accident progression and severe accident phenomena in the spent fuel pool. Severe accident phenomena to consider in this analysis includes heat transfer within the pool, fuel racks, and to surrounding walls, fuel and cladding behaviour ~~(fuel burnup, decay heat, cladding behaviour, etc.)~~, ~~fuel assembly and rack~~ interactions after fuel degradation (zirconium water ~~clad~~ reaction and hydrogen generation, zirconium fire, and corium–concrete interaction), and fission product transport. Such calculations should provide information on the fraction of the fuel assemblies that would be damaged depending on the fuel assemblies arrangement, burn-up and storage time in the spent fuel pool.	Fuel burn-up and "decay heat" are the analysis's input, not output. The words “fuel assembly and rack degradation” should be changed for consistency with the following blankets. “Zirconium clad reaction” should be changed to “zirconium water reaction” for better understanding.		X...and cladding behaviour (fuel burnup, decay heat, cladding behaviour, etc.), fuel assembly and rack degradation including interactions (e.g. zirconium clad and water reaction and hydrogen generation, zirconium fire, and corium–concrete interaction), and fission		For better reading.
Bulgaria	53	13,9	The boundary conditions should define the amount of fuel that is normally replaced during a refuelling outage and a full core unload (if it is prescribed by the operating procedures) to be considered in the calculations	The statement is valid but somehow is not related to PSA elements that actually can define these boundary and initial conditions (BIC). This statement should be related to PDS characteristics, since the BIC should be defined from PDS analysis and APET structure.		X… 13.9. The boundary conditions should be defined in accordance with the PDS as stated in para 13.5. In particular, care should be taken to define the amount of fuel that is normally replaced during a refuelling outage and a full core unload (if it is prescribed by the operating procedures and within the scope of the plant operational states included in the PSA) to be considered in the calculations.		Text modified to provide the relation to the PDS.
ENISS	12	13,12	Delete: (f) Failure of all installed equipment, which may force the operators to decide where to prioritize the use of any remaining non-permanent equipment. OR ~~Failure of all installed equipment, which may force the operators to decide where to prioritize the use of any remaining~~ Use of non-permanent equipment.	This is a deterministic assumption that may have no probabilistic reasoning. Applying this requirement would also mean that non-permanent equipment may not be credited as they are connected to permanent equipment, that has failed (*Failure of all installed equipment..*). If non-permanent equipment is used it may be included in PSA model and it is not necessary to mention it here or at least it is necessary to remove overtight requirements.		X...(f)Subsequent independent failure of installed equipment required to ensure key safety functions for both the reactor and the spent fuel pool (e.g. residual heat removal), which may force the operators to decide where to prioritize the use of any remaining non-permanent equipment.		To consider the use of non-permanent equipment.
Finland	20	13,12	Delete: (f) Failure of all installed equipment, which may force the operators to decide where to prioritize the use of any remaining non-permanent equipment. OR ~~Failure of all installed equipment, which may force the operators to decide where to prioritize the use of any remaining~~ Use of non-permanent equipment.	This is a deterministic assumption that may have no probabilistic reasoning. Applying this requirement would also mean that non-permanent equipment may not be credited as they are connected to permanent equipment, that has failed (*Failure of all installed equipment..*). If non-permanent equipment is used it may be included in PSA model and it is not necessary to mention it here or at least it is necessary to remove overtight requirements.		X...(f)Subsequent independent failure of installed equipment required to ensure key safety functions for both the reactor and the spent fuel pool (e.g. residual heat removal), which may force the operators to decide where to prioritize the use of any remaining non-permanent equipment.		To take account of non-permanent equipment.
Germany	55	13,12	Impact of the accident management strategies for the reactor core to the spent fuel pool (e.g. if the spent fuel pool is located inside the reactor containment ~~building~~, actuation of the filtered containment venting system will lead to more …	For more clearness: Impact of the venting is only existent when the SFP is located inside the containment. If it is located inside the reactor building but not inside the containment (often to be recognized for BWRs) no impact of the venting to an evaporated boiling can be seen.	X			The term "containment building" has been updated in the document to "containment" to comply with terminology in SSG-53.
Germany	56	13,12	Hydrogen release that could result in deflagration~~/~~ or detonation events that fail structures or electrical and/or mechanical equipment;	Editorial	X			
Saudi Arabia	71	13,12	Impact of the accident management strategies for the reactor ~~core~~ to the spent fuel pool […]	“core’ is not necessary		X...pool Level 2 PSA study can be performed separate or combined with Level 2 PSA for the reactor ~~core~~, depending on		
Germany	57	13,13	If the spent fuel pool is located inside the reactor containment building, accident progression analysis should address the impact from combined reactor and spent fuel pool accident on conditions inside and outside ~~in~~ the containment (e.g. pressure, temperature, corium spreading, inflammable gas).	For more clearness: The possible locations of the SFP inside the containment (often for PWR) and inside reactor building but outside containment (often for BWRs) should be considered.	X			
Saudi Arabia	72	13,16	Under dry conditions in the spent fuel pool, the risk of zirconium fire ~~(i.e. water cooled~~ reactors) in water cooled reactors and its propagation should be considered.	Better formulation. In addition, ‘dry conditions in the spent fuel pool’ is not clear.		X...13.16. For fuel utilizing zirconium as fuel claddingUnder dry conditions in the spent fuel pool, the risk of zirconium fire (i.e. water cooled reactors) and its propagation under dry conditions in the spent fuel pool should be considered.		Better formulation.
Saudi Arabia	73	13,17	Consider using ‘spent fuel pool’ or introduce the meaning of SFP before in the text.	SFP abbreviation not defined before.	X			Together with comment 6 from Germany
Finland	21	13,18	For a spent fuel pool located inside a robust building, the overpressure created by steam and heat should be considered when crediting retention of fission products inside the building.	Heat should be mentioned as well as it has significant impact to pressure. Heat may be created for example by steaming, hydrogen generation or burning of hydrogen or zirconium.		X… 13.18. For a SFP located inside a building capable to ensure the confinement function, the overpressure created by steam and heat should be considered when crediting retention of fission products inside the building.		To be in accordance with text in para 13.2.

Israel	5	13,19	In this subsection titled Analyses of Accidents During Fuel Transfer Operations Between The Reactor And The Spent Fuel Pool, it seems that it would be more appropriate for the text in paragraph 13.19 to be: accidents during fuel transfer operations between the reactor and the spent fuel pool (and not between the spent fuel pool and the reactor).		X			
Bulgaria	54	13,23	13.23-13.27: Additional information is required.	No information is provided explicitly about simultaneous consequences mentioned in para. 2.11. Add additional paragraphs about simultaneous consequences.		X… 13.25. In case of severe accident in the reactor concurrent with significant fuel damage located in the spent fuel pool, the source term evaluation should consider the different timing of radioactive releases from those two sources.		New para created.
Bulgaria	55	13,26	A dedicated source term analysis should be performed for the spent fuel pool and for accidents involving fuel assemblies transfer, based on the age distribution of the fuel elements	The statement is not very clear. What transfers are meant here? From reactor to SFP or from SFP to Wet Fuel Storage or? Please explain age distribution of fuel elements.		X… fuel assemblies transfer between the reactor and the spent fuel pool, based on the age distributionburnup of the fuel elements.		It is related to fuel transfer from the reactor core to the spent fuel pool and vice-versa during the modes of normal operation of reactor refuelling.
Saudi Arabia	74	13,26	The ~~core~~ *fuel* inventory of the spent fuel pool at potential accident times should be analysed considering the history of refuelling and the subsequent mixture of newer and older fuel elements	More clarity as 'core' is confusing.	X			
Saudi Arabia	75	13,27	Similar to the reactor *Level 2* PSA, release categories can be defined for the spent fuel pool to […]	More precise formulation.	X			
Saudi Arabia	76	13,27	In addition, the PSA models for the reactor and for ~~fuel in~~ the spent fuel pool should be integrated to correctly model dependencies of the shared systems.	For more clarity.		X...the spent fuel pool Level 2 PSA. In addition, the Level 2 PSA models for the reactor and for ~~fuel in~~ the spent fuel		Better formulation.
China	24	14		It is suggested to supplement the analysis of the influence of the combination of plant operation states of multiple units on level 2 PSA.		X ...in the single unit Level 2 PSA. Some examples might be: ... 14.3 (c)Plant operational states of each unit on the same site (see Section 5 for considerations of other operational states than full power);		Text added to make reference to Section 5 where recommendations related to other operational states are provided.
China	25	14	Consideration for multiple modules is recommended to add into Section 14.	In developing multiple unit accident progression event tree, the interaction of the models for multiple unit should be highly concerned.		X 14.4...When a single reactor unit includes multiple reactor modules, the Level 2 PSA for that reactor unit should consider interactions and dependencies among the reactor modules similarly as reactor units in the Level 2 PSA for the multi-unit nuclear power plant.		Text added to consider multi-modules in a single reactor unit and footnote with referenced added.
ENISS	13	14,1	Paragraphs 14.2 to 14.31 aim at providing recommendations for the development of Level 2 PSA for sites where several units are located, ~~given that~~ suitable for use where national regulatory requirements compel such studies. Given the complexity of models, and the high level of associated uncertainties, the development of such Level 2 PSA is not yet a common practice among the Member States, but it can present an interest to capture some risks relevant to the whole site as well as dependencies among units from the Level 2 PSA perspective, if they were not already addressed in the development of the PSA model for each single unit. ~~Therefore~~ Upon these assumptions, the recommendations in this Section are intended to harmonize approaches in ~~the development of such studies among the~~ Member States which are developing such studies. More information on Member States' experience, practical case studies and guidance on PSA for multiple unit nuclear power plants are provided in Ref. [65]	Text improvement to avoid suggesting that all national regulatory requirements compel such studies. Proposed addition to explain why this practice is not common. Text improvement	X			
Germany	58	14,6	Recommendations provided in paras 4.11-4.18 related to plant familiarization for the development of a single unit Level 2 PSA …	Insertion of a missing paragraph number.	X			
Saudi Arabia	77	14,6	Recommendations provided *in para. 4.18* related to plant familiarization for the development of a single unit Level 2 PSA are also applicable as prerequisites ~~for~~ *to* the Level 2 PSA for multiple unit nuclear power plants.	There is only paragraph 4.18. Editorial: 'for' is replaced by 'to' in order to avoid repetition.		X...14.6. Recommendations provided in paras 4.11-4.18 related to plant familiarization for the development of a single unit Level 2 PSA are also applicable as prerequisites for the development of the Level 2 PSA for multiple unit nuclear power plants.		Reference provided.
Bulgaria	56	14,8	RISK METRICS FOR LEVEL 2 PSA FOR A MULTIPLE UNIT NUCLEAR POWER PLANT	The section is undeveloped. See the para. 11.5 in SSG-3 (rev.1). Similar approach as in SSG-3 should be used relating to LERF and LRF metrics at least. The risk metrics if L2 PSA can be taken from SRS-110 (section 2.3.2)		X… More information on Member States' experience, practical case studies and guidance on PSA for multiple unit nuclear power plants are provided in Ref. [65] and [66][65].		Referen to the MUPSA reports is added.
Canada	11	14,10	In support of this guidance, please also refer to: Safety Report Series No. 96 (2019) -Technical Approach to Probabilistic Safety Assessment for Multiple Reactor Units	This is a suitable reference for the PSA for multiple reactor units on the same site	X			Reference added as Ref. 66. in 14.1

Germany	59	14,12	… Additional accident progression analyses may be needed depending on the differences of reactor technologies ~~/~~ and designs on the site and the identified topics of interest.	Editorial	X			
China	26	14,22	Consideration may be made to simplifying the single unit models definitions if it acceptable.	Simplifying models requires careful consideration because it may neglect some risk contributor sometimes. It is not necessary, so suggest replace "should" as "may" .		X…consideration should be made to simplifying the single unit models before combination with account taken of major risk contributors. Since		Text added to take account of risk contributors in the simplification process.
ENISS	14	14,22	Since the number of units can add significant complexity and size to the accident progression event tree, consideration should be made to simplifying the single unit models before combination. Since each Level 1 sequence results in multiple Level 2 sequences by definition, it is prudent to simplify where possible. Methods to simplify the modelling could include but not be limited to the justified removal of low-risk initiating event contributors, focus first on those initiating events that could affect several units at the site at the same time (e.g. total loss of external power supply, total loss of ultimate heat sink, external flooding, earthquake), the grouping of similar Level 1 sequences under a single PDS and/or grouping release categories to capture the generic representation of an accident sequence. According to the topics of interest identified (see para 14.5), an approach based on the post-processing of the single unit L2 PSA results could be sufficient to obtain relevant insights.	An alternative to the complex combination of single unit models should be mentioned. This approach was used for the Large release frequency assessment for the NUSCALE multi-unit PSA (see chapter 19.1.7 of the NUSCALE Final Safety Analysis Report available on the NRC website).	X			
ENISS	15	14,29	The integration and quantification process for Level 2 PSA for multiple unit nuclear power plants should be based on the approach used in the single unit Level 2 PSA. In case of coupling PSA models from different units into a single PSA model, the major concern would be additional complexity from the additional event tree end states, release categories and combinations discussed above. It can be expected that quantification will involve additional consolidation and screening to include a manageable set of inputs for Level 2 scenarios that need to account for the effect of multiple units undergoing Level 1 and Level 2 aspects. According to the topics of interest identified (see para 14.5), an approach based on the post-processing of the single unit L2 PSA results could be sufficient to obtain relevant insights.	An alternative to the complex combination of single unit models should be mentioned. This approach was used for the Large release frequency assessment for the NUSCALE multi-unit PSA (see chapter 19.1.7 of the NUSCALE Final Safety Analysis Report available on the NRC website).	X			
Japan	14	15	USE IN DESIGN PROCESS FOR NEW NUCLEAR POWER PLANT 15.X. In the design process of a new nuclear power plant, Level 2 PSA, in combination with Level 1 PSA, will be used in an iteration process to establish well balanced safety design architecture. Selected design basis accidents and design extension conditions, and performance of safety related structures, systems and components will be justified by the results of PSA. 15.Y. For the purpose to contribute to demonstrate 'practical elimination' plant event sequences that could lead to an early radioactive release or a large radioactive release, a scoping study may be made searching for specific severe accident sequences which might not reasonably manageable within the scope of Level 2 PSA. Additional measures or design modification can be made based on the results of Level 1 and Level 2 PSA.	Use in design process for new NPP is missing.			X	This is already covered in paras 15.7, 15.14, 15.21 and 15.22.
Japan	13	15,1	Development of a list of severe accident sequences ~~scenarios~~ to be addressed in the NPP design.	In PSA, term "scenario" in not appropriate.			X	The term "scenario" is PSA, see SSG-3. The term scenario has been defined in this safety guide.
Bulgaria	57	15,2	...In addition, the level of detail of the PSA would need to be greater if it were intended to use the Level 2 PSA model in a risk monitor. The guidance about risk monitor application requirements related to PSA model are presented in SSG3 (rev.1) These requirements are valid for the SSC added in L2 PSA model.	The guidance for risk monitor application is insufficient. Given the fact that risk monitor is already wide used, it is believed that details would be of great benefit. At least a reference to the new SSG-3 should be provided.	X			
Saudi Arabia	78	15,2	For example, the scope and the level of detail of a *Level 2* PSA that was intended to provide an estimate of the large release frequency or the large early release frequency […]	More precise formulation.	X			
Bulgaria	58	15,3	(a) Includes an as comprehensive as possible set of internal initiating events, internal hazards, natural and human induced external hazards~~, and~~ (a) Includes an as comprehensive as pos-sible set of internal initiating events, internal hazards, natural and human induced external hazards~~, and~~ (b) Addresses all plant operational states, including startup and operation at power, low power and all the modes that occur during plant shutdown and refuelling (if not screened out), and. (c)Addresses reactor and SFP sources of radioactivity, including simultaneous fuel damage.	The SFP is missing as part of the scope of Level 1 PSA. It could be a significant dominant in the risk profile (especially for those located outside of the containment).			X	Referring the SSG-3 (Rev.1) where the full scope is defined (see para 3.1). The PSA for the SFP is additional.
USA	5	15,3	Purpose rewording the following sentence: "Since the Level 2 PSA relies on the Level 1 PSA model, this should require that the Level 1 PSA:" As follows: "since the level 2 PSA relies on the Level 1 PSA model, development of a full scope Level 2 PSA requires that the Level 1 PSA:"	This change will improve text clarity. The original text may imply that a full scope Level 1 and Level 2 PSA is required, which contradicts first sentence in this paragraph and other paragraphs in the document.		X… model, development of a full scope Level 2 PSA requires this should require that thea full scope Level 1 PSA, as defined in SSG-3 (Rev.1) [4].		
Bulgaria	59	15,5	~~This~~ The full scope PSA will ensure that the insights	It is not quite clear which scope is meant by "this scope…".	X			

Germany	60	15,10	Any other applicable insights or information (such as ~~include~~ a cost–benefit analysis, remaining lifetime of the plant, inspection findings, operating experience, doses to workers...	Clarification	X			
Germany	62	15,11	Headline before 15.11: COMPARISON OF LEVEL 2 PSA WITH PROBABILISTIC SAFETY GOALS OR CRITERIA ~~OR GOALS~~	Consistency with the text in the following paragraphs; and goals come first	X			
Germany	61	15,13	… Several States have also set similar numerical values which have generally been defined as objectives or targets (see Annex III ~~IV~~).	Correction of a typo in the Annex number. Annex IV does not exist in the Safety Guide.	X			
Israel	10	15,13	The Annex mentioned in this paragraph should be Annex III (not Annex IV).		X			
Finland	22	15,18	sequenceswith >> sequences with	Typo	X			
Saudi Arabia	79	15,18	[…] of *sequences with* the highest risk significance.	Editorial: Space missing between sequences and with.	X			
Bulgaria	60	15,25	15.25-15.26: Change the position of the paragraphs.	The statements are valid for generic L2 PSA APET development. Move these recommendations to section 9.			X	These paras are related to the SAM.
China	27	15,32	15.32-15.34	It is recommended to add methods or practices to how to consider the source term using in emergency preparedness and response for multi-unit site.			X	Since the Level 2 multi-unit PSA is not a general practice, it is not possible to provide recommendations on methods or practices for considering emergency preparedness and response for multi-unit sites. In addition, in the current practice in MUPSA, the emergency preparedness and response is considered in Level 3 PSA.
Bulgaria	61	15,34	Additional information is required.	The recommendation is very scarce and insufficient. It will be valuable if additional recommendations are provided about how to select representative sequence(s), especially in case of more than one unit on the site?			X	Since the Level 2 multi-unit PSA is not a general practice, it is not possible to provide recommendations on methods or practices for considering emergency preparedness and response for multi-unit sites. In addition, in the current practice in MUPSA, the emergency preparedness and response is considered in Level 3 PSA.
China	28	APPENDIX I.	Given that multiple organizations participate in responding to severe accidents, and non-permanent equipment may be required. HRA analysis need to consider the familiarity of multiple organizational personnel, such as technical support center personnel, MCR operators, on-site operators, etc., with the relevant severe accident management guidelines, and the familiarity of non-permanent equipment.	It is recommended to add the differences in training between operation program and SAMG.			X	The text already provided gives information about HRA and the factors to be considered in relation to the different actors in the SAMG. There is no need to specifcy more.
Germany	63	Appendix, heading and paragraph numbering	APPENDIX ~~I.~~ CONSIDERATIONS FOR HUMAN RELIABILITY ASSESSMENT IN A LEVEL 2 PSA	There is only one appendix. Please renumbered the paragraph numbers in the Appendix from A.1 to A.7.	X			
Germany	64	References to main document		Layout and consistency need to be improved	X			To be performed after finalization of revision by technical editors.
ENISS	16	References	[71] ~~Bonelli, V. and J. Enjolras, HAMSTER: Human Action Modelling—Standardized Tool for Editing and Recording, in Reliability, Safety and Hazard Assessment for Risk-Based Technologies. Lecture Notes in Mechanical Engineering, P. Varde, R. Prakash, and G. Vinod, Editors. 2020, Springer: Singapore. https://doi.org/10.1007/978-981-13-9008-1_67.~~ Jean-François Enjolras and Anne Gailleton, HAMSTER - A New EDF HRA-Type C Methodology, NPIC&HMIT 2023 July 15–20, 2023, Knoxville, TN	Update of the reference [71] (EDF comment)	X			
Bulgaria	62	FIG. I–1	The second boxes named "heat and mass flows in reactor coolant system" should be changed to " heat and mass flows in containment".	There are two boxes named "heat and mass flows in reactor coolant system".	X			
Saudi Arabia	80	I-2	Examples of those codes *are* ~~is~~ provided in para I-10.	Editorial.	X			
Bulgaria	63	I-12	Codes for simulation of severe accident progression are the same used for deterministic application. Information about computer codes for severe accident progression and their applicability is presented in [I-38]. [I-38] INTERNATIONAL ATOMIC ENERGY AGENCY, Status and Evaluation of Severe Accident Simulation Codes for Water Cooled Reactors, IAEA-TECDOC-1872, IAEA, Vienna (2019)	Computer codes for severe accident progression should be mentioned as well, since this part is one of the major parts in L2 PSA as oppose to L1 PSA.			X	Previous paras provide information of computer codes for deterministic modelling of severe accident progression. The reference suggested is already presented as [I-27].
Germany	66	I-2	Annex I, I-2, item (1), Line 10, last sentence: … ~~Examples of codes in each of these areas are given in para. I-9.~~ The main features of selected mechanistic codes are briefly described in Annex I to Safety Reports Series No. 56, Approaches and Tools for Severe Accident Analysis for Nuclear Power Plants [I-38].	Not able to find examples of mechanistic codes in para. I-9 nor elsewhere in Annex I. In Annex I to Safety Reports Series No. 56, the main features of the mechanistic codes ATHLET-CD, ICARE/CATHARE and SCDAP/RELAP5 are briefly described. We suggest to insert a reference to this publication here and in the list of references to Annex I as well.	X			
Germany	67	I-2	Annex I, I-2, item (2), Line 23, last sentence … Examples of such comparisons are found in Refs [I-1] and [I-2]. The main features of selected integral codes are briefly described in Annex I to Ref. [I-38].	Please add new last sentence, as in Annex I to Safety Reports Series No. 56, the main features of the integral codes ASTEC, MAAP and MELCOR are briefly described. We suggest to insert a reference to this publication here and in the list of references to Annex I.	X			

Germany	68	I-2	Annex I, I-2, item (3),Line 11, last sentence … ~~Examples of those codes is provided in para. I-10.~~	Not able to find examples of dedicated codes in para. I-10 nor elsewhere in Annex I. Please either insert appropriate references or remove the last sentence from the text.	X			
Germany	69	I-2	Annex I,I-2, item (3): *This Section provides a brief description of some specific codes currently in use for Level 2 PSAs, which deal with most or all of the phenomena shown in Fig. I–1. A list of major mechanistic codes is also included.*	Not able to find such description and associated list of codes in paras I-8 - I-10. We suggest to revise para I-7.	X			
Canada	12	Annex I	"REFERENCES TO ANNEX II"	Typo	X			
Germany	65	Annex I	Annex I Heading: REFERENCES TO ANNEX I~~I~~	Correct numbering	X			
Japan	16	Annex I	Annex I REFERENCES TO ANNEX III: Delete the lists which are not referred in Annex I.			X …sentence added at the end of I-1. References to most common codes are provided from [I-5] to [I-37].		Refereences corresponds to examples of codes.
Saudi Arabia	81	I-6	Between I-6 and I-7 EXAMPLES OF ~~INTEGRAL~~ CODES FOR SEVERE ACCIDENT ANALYSIS	The title is not consistent with the list of code categories below.	X			
China	29	ANNEX II		The sample contents of the main report are from previous version of the guide. It's suggested to adjust the contents to be consistent with the structure of the new version of the guide.	X			Updated schedule proposed
Saudi Arabia	82	Table II-1	Please consider revising the table with a more realistic time schedule and consistency with II-2 (well trained team).	Table II-1 considers 5-month training while II-2 states that it assumes a well-trained team. We should also avoid giving wrong impression for the means (number of qualified analysts) and time necessary to perform a Level 2 PSA.	X			Updated schedule proposed
France	9	Annexe II	TABLE II- 1 Example of plan for performance of Level 2 PSA revise the time schedule: 3 years instead of 1 year ?	This plan suppose a L2 PSA development is possible within 1 years. A more reasonable time-schedule is 3 years starting from zero		X… 30 months		Agreement among experts
China	30	ANNEX	Add the terms	It is recommended to add specialized terms for ease of understanding.			X	There is no new terminlogy used.
Bulgaria	64	Table III–1	**Reference III–4** **~~BNRA (2010) Safety Guide. Probabilistic Safety Analysis of Nuclear Power Plants, (in Bulgarian) https://www.bnra.bg/media/2021/05/2rr-07-2010.pdf~~** **"Regulation on ensuring the safety of nuclear power plants", approved with a CM Letter No.245, dated 21.09.2016, promulgated SG, issue 76/ 30.09.2016, amended, issue 37/4.05.2018** **Large release frequency risk metrics Definition** **~~If for Cs-137 in 30km zone > 30TBq~~** **~~if evacuation ends before 12/24/48 hours~~** **"Large releases" shall mean releases of radioac-tive material to the environment, which necessi-tate off-site protective actions to be implement-ed for protecting people and their application cannot be limited in terms of times and areas.** **Safety goal frequency, 1/r.y.** **~~< 1·10-5 for operated NPP~~** **~~< 1·10-6 for new NPP~~** **Accidents with nuclear fuel melting, resulting in early or large radioactive releases to the envi-ronment shall be practically eliminated.**	The risk metrics are presented in the last revision of "Regulation on ensuring the safety of nuclear power plants", approved with a CM Letter No.245, dated 21.09.2016, promulgated SG, issue 76/ 30.09.2016, amended, issue 37/4.05.2018. Note that L1 and L2 PSA guidance that is referred in this document will be updated in the recent future. So, numerical values for frequency targets and definitions of large and early could be developed and presented eventually.	X			
Canada	13	Table III-1	Replace the current write up for Canada with the following: For Operating NPPs (consistent with INSAG 12): 100 TBq of Cs-137 LRF (a release of more than 100Bq of Cs-137) < 1·105/yr For New NPPs (Reference [III-1]: Large Release: LRF (A release of more than 100TBq of Cs-137 or requiring long-term relocation of the population) < 1·10-6/yr Small Release: SRF (A release of 1000TBq of I-131 or requiring temporary evacuation of the local population) < 1·10-5/yr	Canada safety goals should be correctly reflected in the Table.	X			

USA	6	Table III-1	1.The US NRC does not explicitly use a LRF metric as a safety goal. Please reword the 3rd column entry for USA in Table III-1, as follows: "The US NRC does not use LRF as a safety goal. For new reactor design certification reviews, the NRC defined a CDF goal and a Conditional Containment Failure Probability (CCFP) goal, complemented by a deterministic containment performance goal. The US NRC uses the LRF metric for new reactors of 1E-5 as a screening criterion to inform the staff whether new reactor design applicants are meeting the Commission's expectations for a higher standard of severe accident safety performance and increased margin before exceeding safety limits. The US NRC Commission has not approved a formal definition of a large release or a large release frequency. One informal definition for large release frequency is the frequency of an unmitigated release of airborne fission products from the containment to the environment that is of sufficient magnitude to cause severe health effects, regardless of its timing. New reactors transition from LRF to LERF metric at or before initial fuel load and discontinue regulatory use of LRF thereafter (Reference: SRM-SECY-12-0081, "Risk-Informed Regulatory Framework for New Reactors," U.S Nuclear Regulatory Commission, Washington, DC, 2012, ML12296A158)." 2.Since LRF is not a safety goal, 4th column would be best unfilled. 3.Please replace existing reference III-12 Safety goal policy statement of 1986 with the following more recent reference: SECY-13-0029, "History of the Use and Consideration of the Large Release Frequency Metric", U.S Nuclear Regulatory Commission, Washington, DC, March 22, 2013, (ADAMS Accession No. ML13022A207).		X			
Bulgaria	65	Table III–2	**Reference III–4** **~~BNRA (2010) Safety Guide. Probabilistic Safety Analysis of Nuclear Power Plants, (in Bulgarian) https://www.bnra.bg/media/2021/05/2rr-07-2010.pdf~~** **"Regulation on ensuring the safety of nuclear power plants", approved with a CM Letter No.245, dated 21.09.2016, promulgated SG, issue 76/ 30.09.2016, amended, issue 37/4.05.2018** **LERF risk metrics definition** **~~If for Cs-137 in 30km zone > 30TBq~~** **~~if evacuation ends before 12/24/48 hours~~** **"Large releases" shall mean releases of radioac-tive material to the environment, which necessi-tate off-site protective actions to be implement-ed for protecting people and their application cannot be limited in terms of times and areas.** **"Early releases" shall mean radioactive releases to the environment that would require off-site emergency measures for protection of the pub-lic, which is rendered impossible due to insuffi-cient time to implement them.** **Safety goal frequency, 1/r.y.** **~~< 1·10-5 for operated NPP~~** **~~< 1·10-6 for new NPP~~** **Accidents with nuclear fuel melting, resulting in early or large radioactive releases to the envi-ronment shall be practically eliminated.**	The risk metrics are presented in the last revision of "Regulation on ensuring the safety of nuclear power plants", approved with a CM Letter No.245, dated 21.09.2016, promulgated SG, issue 76/ 30.09.2016, amended, issue 37/4.05.2018. Note that L1 and L2 PSA guidance that is referred in this document will be updated in the recent future. So, numerical values for frequency targets and definitions of large and early could be developed and presented eventually.	X			
Canada	14	Table III-2	The following changes are needed: 1) Add one sentence at the top of the cell: Canada has safety goals for LRF and not for LERF. 2) Replace the paragraph (for new NPP) with the following: (for new NPP) The sum of frequencies of all event sequences that can lead to any release to the environment that requires long-term relocation of the population or a release to the environment of more than 1014 becquerels of Cesium-137 shall be less than 10-6 per reactor year.	To provide correct safety goals definition for LRF in Canada.	X			Row deleted
Canada	15	References to Annex III	References to Annex III: [III-1] "Canadian Nuclear Safety Commission (CNSC), Regulatory Document RD-337 REGDOC-2.5.2: Design of New Nuclear Power Plants, RD-337: Design of New Nuclear Power Plants - Canadian Nuclear Safety Commission Ottawa, (2014) "Physical Design: Design of Reactor Facilities, Version 2.1", May 2023."	The old reference RD-337 has been superseded by REGDOC-2.5.2.	X			
Saudi Arabia	83	III-1	*Large* release frequency should be used as an integral indicator of the risk profile covering early and late radioactive releases.	Editorial.	X			
Saudi Arabia	84	Table III-1 and Table III-2.	Table III-1 and Table III-2. These tables need to be revised to provide clear and useful information on large and early release frequency risk metrics and safety goals.	The information provided in the 3rd and 4th columns is often not clear (e.g. for Bulgaria, Czech Republic, Switzerland).			X	This table presents current status of definition of Level 2 PSA goals in Member States with references.

Pakistan	9	ANNEX III TABLE III–2. Examples of Member States practice on LERF definition "S#: Pakistan"	The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated.	Regulation 27(6) of PNRA Regulation on the Safety of Nuclear Power Plant Design - (PAK/911) (Rev.2) states the proposed text. This text is added because in the referred Table of DS-528 as various countries have stated their definitions of LERF.	X			
Japan	17	Annex III TABLE III-1.	III-1. Large release frequency and large early release frequency are the most common measures of risk used in Level 2 PSA. In many Member States, numerical values of this type are used as probabilistic safety goals or criteria. For example, Level 2 PSA risk metrics for large early release frequency should provide information with regard to the frequency of the release, on the release category with regard to the main radioactive material in that release category and the notion of the time of the release. large release frequency should be used as an integral indicator of the risk profile covering early and late radioactive releases. Level 2 PSA risk metrics large release frequency should provide information with regard to both the frequency of the release and on the release categories with regard to the main radioactive materials in that release categories integrated over a period of time. The following tables provide examples of large release frequency and large early release frequency values and definitions in some Member States, with the reference from where such information comes from. Since the Member State's regulatory framework and the role of safety goals are different, these frequency values of Member States should not be compared by themselves.	An explanation related to safety goals or criteria to avoid the misunderstanding is desired.			X	The purpose of this table is indeed to allow comparison of LERF and LRF amon Membr States. The text proposed goes against this objective.
Japan	18	Annex III - Table 1	Large Release Frequency(LRF) risk metrics Definition - Taking into account the Tepco's Fukushima Daiichi Nuclear Power Plants accident, it is necessary to incorporate the viewpoint of the environmental contamination by radioactive materials into safety goal, and to keep the impact on the environment as low as possible if accidents occur. - The frequency of accidents in which the release of Cesium-137 exceeds 100 TBq should be reduced to no more than once in one million reactor years except for those caused by terrorist attacks. Safety goal frequency ~~100 TBq of Cs-137~~ ~~≤1·10-6~~	Add a detailed explanation of Japanese practice.	X			
Japan	19	Annex III - Table 1	REFERENCES TO ANNEX III [III-8]. ~~Review guide on effectiveness assessment of Measures to prevent core damage and measures to prevent containment damage for nuclear power reactors. (Established in 2013, revised in 2017) NRA, Japan, 2017.~~ ~~Document No.5 of the meeting of Nuclear Regulation Authority in Japan, 10 April 2013. (In Japanese)~~ Document No.5 of the meeting of Nuclear Regulation Authority in Japan, 10 April 2013. (In Japanese)	Replace to the proper reference.	X			
Germany	70	Annex III, Table III–1, Ukraine	… criterion / goal for existing plants: $< 1\cdot10^{-6}$ 1/r.y.; criterion / goal for new plants: $< 1\cdot10^{-7}$ 1/r.y.	Clarification	X			
Germany	71	Annex III, Table III–1, USA, Ref. [III-12]	… the overall mean frequency of a large release of radioactive materials to the environment from a reactor accident should be less than 1 in ~~1,000,000~~ 1 000 000 per year of reactor operation.	Editorial			X	Para modified.
Germany	72	Annex III, Table III–2, heading	TABLE III–2. Examples of Member States practice on large early release frequency (LERF) definition.	Please introduce the abbreviation 'LERF' when using it for the first time in the text. The abbreviation appears several times in Table III–2.		X...Examples of Member States practice on large early release frequency (LERF) risk metrics / safety goals definition.		In accordance with title for Table III-1.
Germany	73	Annex III, Table III–2, Czech Republic	More than >1% of Cs-137 of the core inventory released to the environment within 10 hours …	Correction of a typo in the radionuclide notation.	X			
Germany	74	Annex III, Table III–2, Finland and List of references to Annex III	Radiation and Nuclear Safety Authority (STUK), Guide VAL.1, Protective actions in a nuclear or radiological emergency, 20 December 2022.	Please add new Ref. [III-22]. The LERF risk metrics definition (see column 3 in Table III–2) refers to the STUK Guide VAL.1. This is a new Guide, which is not referred to in the Finnish Ref. [III-5] (STUK Guide YVL A.7 published in 2019). Thus, we suggest to add it as a second Finnish Ref. [III-22] in Table III–2 and in the list of references to Annex III.	X			
Germany	75	Annex III, Table III–2, Slovak Republic	More than > 1% of Cs-137 released from the core inventory to the environment within 10 hours after the beginning of the initiating event ~~IE~~	1.) Correction of a typo in the radionuclide notation. 2.) The abbreviation 'IE' is nowhere introduced and not further used in the Safety Guide; it should be replaced by the full term.	X			

Germany	76	Referen-ces to Annex III		Layout and consistency need to be improved	X			To be done with Technical editors.
Hungary	10	TABLE III–2	For operating NPPs: Radioactive release in the case of which urgent precautionary measures are required off the site but no sufficient time is available for their introduction.For new NPPs: a) urgent protective measures are required beyond a distance of 800 m from the nuclear reactor OR b) there is a need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor OR c) there is a need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor OR d) there is a need for any long-term restriction on food consumption.	The Hungarian example is incorrect/incomplete. In the case of Hungary LERF is defined differently for operating and new NPPs, the NSC Volume 3 definition only applies for operating NPPs. For new NPPs the definition (technically the negated version of it) is provided in the NSC Volume 3a in regulation 3a.2.4.0700.: a) no urgent protective measures are required beyond a distance of 800 m from the nuclear reactor; b) there is no need for any kind of temporary action, i.e. the temporary evacuation of the population, beyond a distance of 3 km from the nuclear reactor; c) there is no need for any kind of subsequent protective measure, i.e. the final re-settlement of the population, beyond a distance of 800 m from the nuclear reactor; d) there is no need for any long-term restriction on food consumption.	X			