Table of resolution of NUSSC Members' comments for Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, STEP 7 (DS528) NUSSC 55th Meeting

N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
1.	Belgium	1		About ten links to references are incorrect, which appears as "Error! Reference source not found"		X			
2.	UAE	1	Whole document		In relation to the IAEA documents of "DPPDS537-Safety Guide on Safety Demonstration of Innovative Technology in Reactor Designs", it was commented that If there is a proposed or adopted innovative technology associated with new components, systems and human actions that having safety function, the probabilistic risk assessment (PRA or PSA) level 1 or level 2 should be considered to ensure that the available data for the failures is used or to develop a new methodology to estimate the risk associated with new innovative technology. Consistency of IAEA documents with PSA documents: In the current Level 2 document, there is no any indication of innovation part or Artificial Intelligent technology if adopted in future.			X	IAEA Safety Guides are built on international consensus on the best acceptable practices to achieve a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. Therefore, the use of PSA for innovative technologies and particularly Level 2 PSA, where no sufficient knowledge is available (e.g. lack of knowledge related to severe accident phenomena in advanced reactor technologies and designs), needs to be applied carefully considering its limitations. In this Safety Guide it is acknowledged the use of Dynamic PSA (see para 10.17 with reference) as an innovative technique that could be used for some specific studies in the Level 2 PSA development where classical Level 2 PSA will not provide sufficient details. In addition, the recently approved DS523

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									(revision of the Safety Guide on the Development and Application of Level 1 PSA for NPPs) does not include any mention to advanced methods or innovative technologies.
3.	Iran, Islamic Republic of	9	General comment	The application of PSA Level 2 in Design Extension Condition area to be more clarified.	In this draft SG, no explanation is given for Design Extension Condition.			X	The development of Level 2 PSA aims at demonstrating the sufficiency and balance of the design to cope with severe accident conditions and mitigate their consequences. Therefore, the development of Level 2 PSA implicitly considers the safety features and safety systems designed and qualified for design extension conditions with core melting. Examples of paras are 2.2, 5.5, 5.11, and 5.19. A footnote could be added to para 2.2 as: The development of Level 2 PSA implicitly considers the safety features and safety systems designed and qualified for design extension conditions with core melting.
4.	Iran, Islamic Republic of	10	General comment	The risk monitoring to be explained.	In this draft SG, no guidance is given on risk monitoring.			X	For risk monitoring of the plant in operation, the use of Level 2 PSA is not the main objective.

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5.	Iran, Islamic Republic of	11	General comment	The combination of hazards in PSA level 2 to be explained.	In this draft SG, no guidance is given on combinations of hazards.			X	The methodology for considering the combination of hazards in PSA is described in paras 6.4 to 6.27 as part of Level 1 PSA. That methodology is applicable to Level 2 PSA also, therefore it is not repeated here. The recommendations provided in paras 5.16 to 5.23 aims at adding further recommendations in relation to Level 2 PSA. In addition, section 8 provides recommendations related to human and equipment reliability assessment for Level 2. Those recommendations also consider the effects of hazards in the context of Level 2 PSA.
6.	Sweden	31	General	Add list of acronyms and abbreviations				X	The list of abbreviations will be considered according to the IAEA publishing rules.
7.	USA	1	1.5	Existing text: "Thus, a ~~comprehensive~~ probabilistic safety assessment (PSA) is required to be performed 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~~."	Delete qualifier "comprehensive" and delete "in relation to potential internal initiating events and internal and external hazards as well as their combination." The text implies that all external hazards and their combinations have to be assessed by the PSA, i.e. a full-scope PSA is required. In contrary, as further detailed in para 2.24, 15.2, 15.3, 15.4 the guide allows for more limited scope PSA, depending on the		X Thus, a full-scope ~~comprehensive~~ probabilistic safety assessment (PSA) ~~is required to be performed~~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.		The notion of comprehensive changed to full scope to in-line with the scope recommended in DS523 para 2.2. The term "required" is deleted. Modification proposed to comply with later paras.

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					probabilistic safety goals, or for "alternate approaches used to demonstrate the risk from those initiating events and hazards and operating states that are not in the model does not threaten compliance with the probabilistic safety goals or criteria". See also revisions to para 2.33 to match wording in SSG-3, where a full scope PSA is recommended, not required.				
8.	Germany	1	1.6 (1)	In Level 1 PSA, the design and operation of the plant are 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. Level 1 PSA provides insights into the strengths and weaknesses of structures, systems and components (SSCs) important to safety and procedures in place or envisaged as preventing core and/or fuel damage. <u>Further 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].</u>	For consistency with DS523 (Revision of SSG-3), please add a sentence similar to the one in DS523 1.4. (2) as a cross link to the other Guide.	X			
9.	Germany	2	New para 1.7A	<u>Level 1 PSA, Level 2 PSA and Level 3 PSA are sequential analyses, with the results of each assessment usually serving as a basis for the PSA at the next level. Level 1 PSA provides insights into design weaknesses and into ways of preventing accidents leading to core and/or fuel damage, which might be the precursor to accidents leading to major releases of radioactive material with potential consequences for human health and the environment. Level 2 PSA provides insights into the relative importance of</u>	Please add a new para for consistency.Text is taken from DS523 (PSA Level 1) para 1.5 (without one sentence referring to SSG-4).		X In international practice, three sequential levels of PSA are generally recognized:		This is a repetition from previous paragraph 1.6. If, the information related to the sequential aspect is considered essential, that could be added as part of the sentence in 1.6 as proposed.

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				accident sequences leading to core and/or fuel damage in terms of the severity of the releases of radioactive material they might cause, and insights into weaknesses in confinement functions and measures for the mitigation and management of severe accidents, along with ways of improving them. Level 3 PSA provides insights into the relative importance of accident prevention and mitigation measures, expressed in terms of adverse consequences for the health of both plant workers and the public, and the contamination of land, air, water and foodstuffs. In addition, Level 3 PSA provides insights into the relative effectiveness of aspects of accident management relating to emergency preparedness and response.					
10.	Germany	3	1.8 (a)	~~Level 1 PSA provides information on the accident sequences that lead to fuel damage and hence provides the starting point for Level 2 PSA.~~ The accident sequences identified by Level 1 PSA may not include information on the status of the SSCs dedicated to ensuring the confinement function (e.g. the containment systems in pressurized water reactors) that mitigate the effects of severe accidents.	Please delete this sentence.			X	Level 1 PSA indeed provides information on the accident sequences that lead to fuel damage (core or spent fuel pool) which are the input for the development of Level 2 PSA.
11.	Russian Federation	1	1.8(c), second sentence.	Paragraph 9.2 explains the difference between accident progression event trees and containment event trees	It is proposed to exclude (reasons - see comments to Item 9.2).		X Para 1.8 (c) modified as: An accident progression event tree (APET) is used to model accident progression to identify accident sequences that challenge the SSCs dedicated to ensuring the confinement function and lead to releases of radioactive material to the environment. Footnote to 1.8(c) Such event trees are also termed containment event trees. The term		Para 1.8 (c) modified for clarification of the term used in this safety guide. A footnote was also added in relation to the term containment event trees.

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							accident progression event trees has been chosen throughout this safety guide, like in the ASAMPSA2 project [21], because it is more generally applicable. In addition, Para 9.2 was modified accordingly: "In Level 2 PSAs, event trees are used to delineate the sequence of events and severe accident phenomena after the onset of core damage that challenge containment integrity and the successive barriers to radioactive material release. They provide a structured approach for the systematic evaluation of the capability of a plant to cope with severe accidents. Their use is shown in Fig. 1. Such event trees, termed accident progression event trees (APET) in this guide, include modelling of phenomena, systems actuation or failure, human actions and all impacts on the confinement of radioactive products or the radioactive releases in the environment.		
12.	Japan	1	1.8(d)	Source term analysis is used to determine the quantities and timings of radioactive material released to the environment from each of the release categories.	Not only quantities of radioactive material release, but also timings of release are needed to analyze.	X			
13.	USA	2	1.11	Level 1 and Level 2 PSA, **of varying scope and level of detail**, have been performed for almost all power plants.	As implied further in this safety guide, a full-scope Level 2 PSA may not be required, depending on the objectives described in paras 2.3 and 2.10. additionally, some PSAs use the LERF metric, and external hazards PSA is not required, see para 2.10.	X			
14.	Egypt	1	1.16	Although the recommendations provided in this Safety Guide are intended to	The word "inclusive" misleading, it means that the guide includes all NPP			X	PSA is a in general a technology neutral

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				reflect a technology independent methodology, ...	technology, this guide should be technonlgy neutral.				methodology, considering the meaning of technology neutral as it does not provide any recommendation technology related. On the contrary, the Level 2 PSA methodology described in this safety guide (as well as the one in SSG-3), explicitly considers the safety features and safety systems present only in NPPs, e.g. containment safety features, to manage severe nuclear accidents. Therefore, the methodology is technology inclusive.
15.	Germany	4	1.17	This Safety Guide addresses the necessary ~~methodological~~ technical features of Level 2 PSA for nuclear power plants (both existing and new plants), on the basis of internationally recognized good practice in relation to its application, with an emphasis on the procedural steps and essential elements of the PSA rather than on details of the modelling methods. This Safety Guide includes all the steps in the Level 2 PSA process, up to and including the determination of the detailed source terms needed as input into a Level 3 PSA.	Please put in line with DS523			X	It is important to highlight the "methodology" part in Level 2 PSA rather than in Level 1 PSA which is more straight forward. Therefore, it does not need to be quoted as in Level 1 PSA. All safety standards are drafted and approved based on the international consensus on the best good practices. This type of text is always presented in the foreword of all safety standard. Therefore, there is no need to repeat it here.

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16.	Germany	5	1.18	~~This Safety Guide describes all aspects of the Level 2 PSA that need to be carried out if the starting point is a full scope Level 1 PSA as described in SSG-3 (Rev. 1) [4].~~ The scope of a Level 2 PSA addressed in this Safety Guide includes all operating states of the plant (i.e. in power operation and shutdown) and all potential initiating events and potential hazards, namely: (a) internal initiating events caused by random component failures and human error, (b) internal hazards and (c) external hazards, both natural and human induced, as well as combinations of hazards, such as consequential (subsequent) events, correlated events and unrelated (independent) events addressed in a full scope Level 1 PSA as described in SSG-3 (Rev. 1) [4]. If the objectives of the Level 2 PSA are limited, only the relevant recommendations provided in this Safety Guide apply; if the scope of the Level 1 PSA is limited (see paras 2.8-2.9), additional analysis to that described in this Safety Guide may need to be carried out.	Please extend the Scope to be in line with DS523 para 1.11.		X1.18. The scope of a Level 2 PSA addressed in this Safety Guide includes all modes of normal operation of the plant (i.e. startup, power operation, shutting down, shutdown, maintenance, testing and refuelling) and considers the Level 1 PSA results obtained for all potential initiating events and potential hazards, (i.e. a full scope Level 1 PSA as described in SSG-3 (Rev. 1) [4]), namely: (a) internal initiating events caused by random component failures and human error, (b) internal hazards and (c) external hazards, both natural and human induced, as well as combinations of hazards, such as consequential (subsequent) events, correlated events and unrelated (independent).		Text updated regarding the terminology in IAEA Safety and Security Glossary which defines normal operation state and the different modes as presented. Level 2 PSA does not look at internal initiating events but to plant damage states which are a group of end states coming from several internal initiating events, internal hazards and external hazards. Text updated to comply with the development of Level 2 PSA as stated in para 1.6.
17.	Germany	6	1.19	If the aim of the PSA is to determine all the contributions to risk to public health and society, then the PSA will need to take into account in the calculation of the source term the potential for release from other sources of radioactivity from the plant, such as irradiated fuel and stored radioactive waste. 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.	Please add a MUPSA sentence - in line with DS523, para 1.12 - which may be important for SMRs.		X This Safety Guide also covers the development of Level 2 PSA for sites where several units and spent fuel pools are located, which may be considered given that national regulatory requirements compel such studies, as part of the quantification of the source term at the site level.		Terminology adapted from technical editors. The purpose is not to quantify risk metrics at the site since this is a national requirement, but the source terms at the site, which includes all potential sources of radioactive releases.

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				This Safety Guide also covers multi-unit aspects, which may be considered when developing a Level 2 multi-unit PSA to quantify multi-unit risk metrics.					
18.	Egypt	2	1.20	The recommendations provided in this Safety Guide are intended to be technology independent to the extent possible.	The word "inclusive" misleading, it means that the guide includes all NPP technology, this guide should be technonlgy neutral.			X	See answer comment 14.
19.	Finland	1	Section 2 or 15 (scope of level 2 PSA)	Add a reminder that a design phase level 2 PSA should be sufficiently detailed to facilitate the identification of need for design improvements. The properties of the design (e.g. the confinement function) with regard to severe accident prevention and mitigation are decided during the design phase. It can be expensive or even impossible to implement good design improvements later.	This is a complementary requirement to requirements given in SSG-3 (draft) section for "Use of PSA for design evaluation". A specific feature of level 2 PSA is that it addresses SSCs (especially the confinement function) which are difficult or impossible to fix later if weaknesses are identified.		X Para 2.4 modified as:2.4… In particular for the design stage, the detail of Level 2 PSA should be sufficient to achieve the above mentioned objectives considering the difficulty or impossibility to implement design safety features to manage severe accidents in a later stage.		
20.	Ukraine	6	para 2.1 line 1	Incorrect reference to GSR Part 4 should be changed to [2]	Editorial	X			
21.	USA	3	2.2	… and, **for new designs**, contribute to demonstrating the "practical elimination of plant event sequences"	Practical elimination applies to new designs as described in reference [9] and para 2.3.f.		X … and, for new reactor designs, contribute to…		The comply with 2.3 (f).
22.	WNA	1	2.2	IAEA Safety Standards Series No. SSG-88, …	Comment: Be aware that the deliverable is still unpublished and apparently there is a lack of consensus concerning its content.			X	The draft SSG-88 was currently approved by CSS.
23.	WNA	2	2.2	The provisions to manage severe accidents…	Here it is interesting to point out that the notion of "**provision**" covers both material and immaterial elements of what I call the "safety architecture" of the installation.	X			Term "provision" changed to "safety provision" to comply with the definition used in the safety guide (now footnote 3, before footnote 5)

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24.	WNA	3	2.2	(a) To gain insights into the progression of severe accidents and the performance of the confinement function ensured by dedicated SSCs (e.g., the containment);	A first comment concerns the fact that we systematically refer to SSCs but, more generally, it would be appropriate to speak of safety architecture and the provisions of all kinds that make it up, both material and immaterial: SSCs, characteristics intrinsic; procedures, etc. Each of these provisions should be characterized by its physical performance, i.e., the ability to carry out the requested mission, and the reliability that characterizes its intervention. Such a paradigm shift would allow easier integration of innovative solutions in the safety analysis as well as the intercomparison of facilities with different technologies. In other words, this paradigm shift seems essential to move towards the harmonization of safety approaches for the design and assessment of innovative installations.			X	Wrong para 2.2, actually para 2.3. There is no contradiction in the definition of safety provisions and safety architecture. They are equivalent, but the term structures, system and components (SSC) is recognized in the glossary as the appropriate terminology. Therefore, there is no need to change to safety architecture.
25.	WNA	4	2.2	(b) To provide an input into the development of plant specific accident management guidance and strategies;To provide an input into determining plant specific options with regard to design and accident management guidelines and strategies aiming to risk reduction;	The advantage of having an unambiguous representation of the safety architecture and the provisions that make it up would make it easier to meet these two objectives.			X	Wrong para 2.2, actually para 2.3See answer to comment 24.
26.	WNA	5	2.2	(c) For new reactor designs, to demonstrate the 'practical elimination' of plant event sequences that could lead to an early radioactive release or a large radioactive release.	Cf. the previous comment concerning the notion of "safety architecture".			X	Wrong para 2.2, actually para 2.3See answer to comment 24.

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27.	France	1	2.3	The objectives of Level 2 PSA should be defined. These can include the following:…(l) For new reactor designs, **to contribute** to demonstrate**ing** the 'practical elimination' of plant event sequences that could…	For consistency with SSR-2/1, DS508, SSG-88 (DS548) and §2.2 of this DS528	X			
28.	Iran, Islamic Republic of	2	2.3	(a) To gain insights into the progression of severe accidents, the performance of the confinement function and minimizing release of radioactive material ensured by dedicated SSCs;	Minimizing releases of radioactive substances shall be considered as a goal of severe accident management. It is not all about the confinement function. For instance, the operator should establish filtered containment venting in some cases to prevent large release;		X(a) To gain insights into the progression of severe accidents and the performance of the confinement function, ensured by dedicated SSCs (e.g. the containment), to minimize the release of radioactive material;		Sentence modified for better reading.
29.	Iran, Islamic Republic of	3	2.3	The objectives and applications of Level 2 PSA should be defined. These can include the following:	Some of the items mentioned are the applications of the PSA level 2 rather than the objectives.			X	The list in para 2.3 are objectives. There are applications that allow to achieve the objectives, but they are not mentioned here.
30.	Iran, Islamic Republic of	4	2.3	To provide an input into determining plant specific ~~options~~ capabilities with regard to design and accident management guidelines and strategies aiming to risk reduction;	It is more common to use plant capabilities instead of plant options.			X	Item (h) aims at design stage, where design options are explored. Plant capabilities are considered in (a), (b), (e), (g), (i) and (l).
31.	Iran, Islamic Republic of	5	2.3	(m) To gain insights into the cliff edge effects	Level 2 PSA can provide insights into possible cliff-edge effects, and to ensure that the residual risk accrued after the mission time is negligible.		X (m) To gain insights into possible cliff edge effects leading to radioactive releases.		Even though it could be understood that the cliff edge effect (as a possible failure mode) is covered by (e), the additional point is added. However, the relation with cliff edge effects and radioactive releases

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									needs to be explicitly mentioned.
32.	Iran, Islamic Republic of	6	2.3	(n) to evaluate plant modifications related to severe accident and optimize safety decision making processes applied both by utilities and regulatory bodies	Level 2 PSA should be updated as changes are made to the design or operation of the plant.			X	Items (c), (g) and (h) already covers the proposed comment.
33.	Russian Federation	4	2.3	(m) To provide base list of representative sever accidents for deterministic analysis.	Ensuring the completeness of the objectives of the Level 2 PSA. As a objectives of level 2 PSA, it is proposed to add in paragraph 2.3: "To provide base list of representative sever accidents for deterministic analysis".			X	The list of representative severe accidents for deterministic analysis is already covered by the 2.3 (a).
34.	USA	4	2.3 (d)	**Most common** ~~typically, such~~ probabilistic safety goals or criteria related to large release frequencies and/or large early release frequencies**, as further described in para 2.16;**	Referencing para 2.16 to clarify probabilistic safety goals	X			
35.	ENISS	1	2.6	"In undertaking a Level 2 PSA, there are two types of approaches likely to be encountered depending on the overall objective of the PSA project and the software capabilities for developing the probabilistic models. The first is a separated approach, whereby the Level 2 PSA aims to extend an existing Level 1 PSA (as described in para 1.6) and is developed in a different computer tool than the one used for Level 1 PSA. The second is an integrated approach, whereby the Level 2 PSA is part of an integrated Level 1–Level 2 PSA with the use of the same computer tool. The integrated approach has mainly been applied to the latest Level 2 PSA developments for new nuclear power plants equipped with water cooled reactors, but also as an alternative	Paras 2.6, 5.9 and 9.1 use the terms "integrated approach" and "separated approach" but the definitions seem to differ (or current wording may be ambiguous):- in paras 5.9 and 9.1, these approaches seem to be defined according to a "tool orientation": integrated approach refers to a linked event tree or linked fault tree approach, where L1 and L2 PSA are combined in a single computer tool and a same database is used, in contrast to the separated approach.- in paras 2.6, these approaches seem to be defined according to a "project management orientation": separated approach seems to refer to a construction of a L2 PSA "after the Level 1 PSA is complete" while an integrated approach takes into		X Paras 2.6, 5.9 and 9.1 were modified to ensure consistency as:2.6… Para 2.6: In undertaking a Level 2 PSA, there are two types of approaches likely to be encountered depending on the overall objective of the PSA project and the software capabilities for developing the probabilistic models. The first is an integrated approach where the Level 1 and Level 2 PSA models are developed, linked and quantified in a single software tool. 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 /		Relevant paras modified to clarify the choice of using an integrated or a separated approaches for the development of PSA (Level 1 and Level 2). This choice is indeed related to the project management which includes the choice over the computer codes to be used, but it is not the only consideration.

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				for advanced nuclear power plant designs equipped with non-water cooled reactors for which significant core damage is not in the scope of the analysis. 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. 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, **but** 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 have to be considered wherever possible in the Level 1 PSA or Level 1 PSA has to be expanded (see para. 5.6). In an integrated approach, the information from Level 1 PSA that are needed for the Level 2 PSA is implicitly available. In either approach, when linking the Level 1 and Level 2 PSA models, typically via the specification and quantification of PDSs, it should be ensured that the Level 2 PSA takes fully into account the initial and boundary conditions from the Level 1 PSA model and the dependencies between the Level 1 PSA and the Level 2 PSA."	consideration the realization of L2 PSA since the beginning of the project, that may/should have an impact on the scope of the L1 PSA.These definitions are different: it is possible to perform a L2 PSA after L1 PSA (in a sequential manner) but within a single computer tool using a same database.This should be clarified, and the implications addressed. For instance, the para. 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." seems to be in opposition with para 5.9 :"If the Level 1 PSA and the Level 2 PSA are an integrated model developed in a linked event tree or linked fault tree software many of characteristics listed later in paras. 5.10-5.12 will be implicitly available for the Level 2 PSA to use without being made explicit for the PDSs definition".		information / results from Level 1 to Level 2 would be required. ASAMPSA2 provides information on the advantages and disadvantages of each approach [21]. The integrated approach has …. Para 2.9 parenthesis deleted referring to 2.6. Para 5.7 modified as: If the Level 2 PSA is developed as part of an integrated Level 1 – Level 2 PSA (see para 2.6), many of the PDS characteristics listed later in paras. 5.10-5.12 will be implicitly available for the Level 2 PSA model. Such an approach may allow to reduce the number of PDS needed. In any case, even though the structure of the PDSs could be simpler in an integrated Level 1 PSA and Level 2 PSA model, the analyst should verify that simplifications or assumptions in Level 1 PSA model will not screen out possible PDSs contributing to radioactive releases. 5.9… The characteristics specified for the PDSs are generally left to the discretion of the analyst. Examples of characteristics are given in paras. 5.10-5.12. It should be noted that the level of detail of characteristics used to define the PDSs depends on the case used for the development of Level 1 PSA and Level 2 PSA (see para 2.6). If the Level 2 PSA is developed as an extension of Level 1		

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							PSA, the definition and selection of characteristics specified for the PDSs should be justified. (Rest of para deleted) 9.1… For the development of a Level 2 PSA event tree model, two different approaches can be used: an integrated approach and a separated approach as described in para 2.6 ~~which differ mainly by the way information is transmitted from Level 1 PSA to Level 2 PSA (see para 2.6). In an integrated approach, Level 1 and Level 2 models are combined and developed as one study and a single computer code might be used (see para 2.6). In a separated approach, allowing the use of specific computer codes for Level 2 PSA, Level 1 and Level 2 models are separated so that a specific interface has to be defined to ensure the transmission of the necessary information from Level 1 to Level 2 PSA.~~ The Level 2 analyst should…		
36.	WNA	6	2.6	The integrated approach has mainly been applied to the latest Level 2 PSA developments for new nuclear power plants equipped with water cooled reactors, but also as an alternative for advanced nuclear power plant designs equipped with non-water cooled reactors for which significant core damage is not in the scope of the analysis.	With a view to harmonization, this document could propose synonyms for the notion of "core damage" for concepts that do not have a core in the conventional sense of the term, which is the case, for example, of MSR. One could for example evoke a solution of continuity for the mode of attack of the ultimate containment which in the case of conventional reactors is materialized by the contact with the corium and, in		X the term "significant core damage" changed to "significant core degradation" and a footnote 4 was added: The notion of "significant core degradation" for some non-water cooled reactor technologies, which might not have a "reactor core" as it is conventionally understood for water cooled reactors, might not be applicable. However, the analysis		The term "significant core damage" change to "significant core degradation" as in the IAEA Safety Security glossary Ed. 2022. Footnote added.

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					the case of the non-conventional concepts, with a contacting of the heat source (e.g., fuel salt) with containment.		would aim at identifying the challenges to the containment due to contact of the heat source (e.g. fuel salt) and related phenomena that might lead to radioactive releases.		
37.	USA	5	2.7	"the scope of level 2 PSA should be determined by **its defined objectives see para 2.3 and** its specific intended uses **and applications, as further detailed in para 15.2**.	Reference to para 2.3 and 15.2 to explain how the scope of the PSA varies	X			
38.	USA	6	2.12	Add at the end: **As stated in para 1.19, releases from other sources of radioactivity from the plant, such as irradiated fuel and stored radioactive waste" is not detailed in this safety guide.**	Reference para 1.19 for clarity.	X			
39.	ENISS	2	2.13	~~Any analysis and assumptions associated with a Level 2 PSA should be as realistic as possible and include an uncertainties assessment, consistent with the intent of the study being undertaken. The ultimate product of a Level 2 PSA, then, will be a description of a number of challenges to the containment, a description of the possible responses of that containment and an assessment of the consequent releases to the environment and their associated frequencies. The descriptions will include the inventory of material released, its physical and chemical characteristics, and information on the time, energy, duration and location of the releases. Related uncertainties should be part of these descriptions~~	This paragraph has little to do with the "scope of level 2 PSA". It carries too detailed recommendations at this point of the guide and above all the content of this paragraph is already partially integrated in the para. 3.7.		X2.13. Any analysis and assumptions associated with a Level 2 PSA should be as realistic as possible and include an uncertainties assessment, consistent with the intent and scope of the study being undertaken. ~~The ultimate product of a Level 2 PSA, then, will be a description of a number of challenges to the containment, a description of the possible responses of that containment and an assessment of the consequent releases to the environment and their associated frequencies. The descriptions will include the inventory of material released, its physical and chemical characteristics, and information on~~		Text modified to confirm recommendations related to the assumptions and uncertainties depending on the scope and objective of Level 2 PSA. The part of the para covered in 3.7 was deleted. To cover the specifics of the inputs for Level 3 PSA.

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							~~the time, energy, duration and location of the releases. Related uncertainties should be part of these descriptions.~~New para 2.14 added as: 2.14. If the scope of the PSA study considers the Level 3 PSA, the scope of the Level 2 PSA should consider the input requirements needed to conduct the Level 3 PSA.		
40.	USA	7	2.13	Any analysis and assumptions associated with a Level 2 PSA should be "as realistic as possible, **commensurate with the intended uses and applications of the Level 2 PSA.**	Degree of realism should be dictated by the intended use. As realistic as possible may involve significant PSA development effort, not always justified.	X			
41.	Germany	7	Heading after 2.14	PROBABILISTIC SAFTY GOALS OR ~~REFERENCE VALUES AND~~ RISK METRICS FOR LEVEL 2 PSA	For consistency with DS523		X REFERENCE VALUES, PROBABILISTIC SAFETY GOALS OR CRITERIA AND RISK METRICS FOR LEVEL 2 PSA Para 2.16 modified as:2.16. The general recommendations related to reference values, probabilistic safety goals or criteria and risk metrics used in PSA presented in paras 2.10–2.15 of SSG-3 (Rev. 1) [4] are applicable to Level 2 PSA…		In fact, to be consistent with the text in DS523 para 2.10-2.15, the title in DS523 should also mention "reference values". Some member states use "reference values", other "probabilistic safety goals" and others "Probabilistic safety criteria" as mentioned in DS523.
42.	WNA	7	2.20	In the operating lifetime of a nuclear power plant, modifications are often made to the SSC design or to the way the plant is operated.	Here also it would be interesting to homogenize the SSCs and the other components of the "safety architecture" through the wording "provision".			X	Safety provisions are design provisions covering the design of SSC and those procedures specific for the operation of those SSCs required during severe accident. Here it is

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									more general, and it is better to specified both the design of SSC and the change in the operating procedures.
43.	WNA	8	2.20	Additional statistical data on the frequencies of initiating events, the probabilities of component failure	The notion of "provision" will allow considering the probability of failure on an immaterial provision (e.g., a procedure).			X	See answer to comment 42. In addition, failure to apply a procedure is not considered as part of the term "safety provisions".
44.	WNA	9	2.20	A PSA that undergoes periodical updating is termed a 'living PSA'. The updating of a PSA should be initiated by a specified process, and the status of the PSA should be reviewed regularly to ensure that it is maintained as a representative model of the plant and is fit for purpose.	The availability of the "plant safety architecture" will allow to ease the consideration of the plant modifications.			X	Consideration of any plant modification should go through a safety assessment process, which covers the design of SSCs, the operating procedures, the emergency operating procedures, as well as maintenance, texting and in service inspection activities, and relevant radiation protection considerations for their implementation and more. Keeping an updated model of the plant for the purpose of PSA calculations (i.e. living PSA) and using it at the design stage for the modification has the advantage to obtain risk insights related to that modification. This is

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									only what the text intends to highlight.
45.	Sweden	1	2.24	…PSA. Quantitative results…….	Editorial, new sentence	X			
46.	WNA	10	2.24	Therefore, in order to use the PSA results for the verification of compliance with existing probabilistic safety goals or criteria, a full scope PSA involving a comprehensive list of initiating events and hazards and all plant operational states should be performed unless the probabilistic safety goals or criteria are formulated to specify a PSA of limited scope, or alternative approaches are used to demonstrate that the risk from those initiating events and hazards and operating states that are not in the model does not threaten compliance with the probabilistic safety goals or criteria.	The PSA should also be used to assess the degree of progressiveness in the course of the accident sequence to ensure that there will not be excessive discontinuities in terms of consequences, but for this it would be interesting to explicitly link the PSA type analysis with the structure of the defense in depth which is put in place and its different levels. Here again, the notion of safety architecture could be useful to structure the approach.			X	Classical PSA is a snapshot in the progression of accident. Advanced PSA methods, such as Dynamic PSA, are able to cover the discontinuities in the progression of the accident. Here, the text aims at recommending the scope needed to use PSA results for comparison with probabilistic safety goals or criteria, if set.
47.	Ukraine	8	2.30	Incorrect references to ~~2.19~~2.19–2.22	Editorial	X			
48.	Sweden	2	2.30	Strange cross reference " paras. 2.192.19-2.22" need to be corrected	Editorial	X			
49.	WNA	11	2.30	The PSA should address the actual design or, in the case of a plant under construction or modification, the intended design or operation of the plant as part of the periodic safety reviews, which should be clearly identified as the basis for the analysis.	It would be extremely useful for that to have a "living" representation of the safety architecture.			X	See answers to comment 44.
50.	Russian Federation	5	2.23-2.34	No	Paragraphs 2.23 -2.34 under the heading of the guide "USE OF PSA IN THE DECISION MAKING PROCESS" look "superfluous" in this Level 2 PSA guide.			X	IAEA safety standards provide recommendations of what should be done to achieve and maintain a

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									high level of safety. Recommendations in IAEA safety guides are not meant to provide how these recommendations are actually implemented. The text presented in the paras mentioned are similar to those from previous versions of IAEA safety standards as well as on recently approved IAEA safety standards.
51.	WNA	12	2.31	In this case, the insights gained from PSA should be considered in combination with the insights gained from the assessment of engineering safety features and deterministic safety analysis to make decisions about the safety of the plant.	Here also insights from the assessment of immaterial provisions (such as procedures or inherent characteristics) should be considered for the living PSA and the design process.			X	The text recognises the advantages and limitations of PSA and that is why it recommends that deterministic safety analyses and the assessment of engineering safety features should also consider insights from PSA.
52.	WNA	13	2.32	This should be done for all probabilistic safety goals or criteria defined for the plant, including those that address system reliability.	As well as the reliability of other immaterial provisions.			X	The concept of reliability is not adequate for operating and emergency procedures, as immaterial provisions, on the contrary they are assessed to be effective and appropriate to operate safely the pant

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									and to manage accident situations to the safe state. In addition, PSA results intrinsically incorporate emergency procedures.
53.	USA	8	2.33	The PSA should **aim** ~~be set out~~ 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.** ~~Screening may be applied to address negligible contributors to risk and focus the study on the most risk significant elements~~. **Such limitations should be acknowledged when using PSA to support decision making. The use of the full scope PSA model is therefore recommended**. ~~If the regulatory standards of a member state require it, a full scope PSA should be conducted.~~	wording should match latest wording in SSG-3, para 2.23. original wording implied all accident sequences should be identified. Wording improved to state non-negligible accident sequences. Secondly, development of a full scope PSA should be recommended, not required.	X			As proposed, it is a repetition of para 2.23 of DS523, therefore it might be deleted or only reference.
54.	Japan	2	2.34	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 severe accident management guidelines strategies and actions. These can be identified by considering the contributions to the risk from groups of initiating events, and the	To clarify that the consideration of costs and benefits is only one aspect.	X			

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				importance measures for SSCs and human errors. Where the results of the PSA indicate that changes could be made to the design or operation of the plant to reduce risk, the changes should be incorporated where reasonably achievable (e.g., taking the relative costs and benefits of any modifications into account).					
55.	Sweden	3	2.34	…guidelines strategies. (remove the words "and actions")	Editorial	X			
56.	WNA	14	2.34	These can be identified by considering the contributions to the risk from groups of initiating events, and the importance measures for SSCs and human errors.	The notion of "importance measure" deserves clarification.		X footnote added as: Typical importance measures used in probabilistic safety assessment are Fussell-Vesely importance, Birnbaum importance, risk reduction worth and risk achievement worth (described in para 5.170 of SSG-3 (Rev. 1) [4]) giving a perspective on how an individual basic event, groups of basic events, credited systems and groups of initiating events contribute to the overall risk profile.		
57.	Egypt	3	3.5	Paragraphs 3.6 - 3.7 provide recommendations on meeting Requirements 1 and 14 GSR Part 4 (Rev.1) [2] in relation to the scope of the Level 2 PSA project.	Paragraphs discuss the scope of the Level 2 PSA project start from: 3.6 to 3.7.	X			
58.	ENISS	3	3.6	The scope of the Level 2 PSA project should be determined by the overall scope of the Level 2 PSA, as described in paras 2.5–2.14, following a graded approach to define the scope and the methods used for modelling the severe accident phenomena and for the contribution of the SSCs to the risk of radioactive releases depending	Paragraph too cumbersome, to rephrase.		X3.6. The scope of the Level 2 PSA project should be determined by the overall scope of the Level 2 PSA, as described in paras 2.5–2.14. The scope of the Level 2 PSA project should , follow~~ing~~ a graded approach to define the scope and the methods used for modelling the		First sentence split to be more readable.

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				on their source (see para 1.19). A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs be part of the installation containing potential sources of radioactive releases other nuclear power plants (e.g. failure tree and event tree development, assumptions related to human reliability analysis or equipment reliability data, fragility curves (if applicable) and reliability of digital instrumentation and control systems, including computer based systems used to control the process in the installation).			severe accident phenomena and for the contribution of the SSCs to the risk of radioactive releases depending on their source (see para 1.19). A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs be part of the installation containing potential sources of radioactive releases other nuclear power plants (e.g. failure tree and event tree development, assumptions related to human reliability analysis or equipment reliability data, fragility curves (if applicable) and reliability of digital instrumentation and control systems, including computer based systems used to control the process in the installation).		
59.	Sweden	4	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 (e.g. fault tree and event tree development…"	Editorial	X			
60.	WNA	15	3.6	A graded approach, for instance, could be applied to the level of detail considered in the probabilistic modelling of SSCs be part of the installation containing potential sources of radioactive releases other nuclear power plants (e.g. failure tree and event tree development, assumptions related to human reliability analysis or equipment reliability data,	It is interesting to note that on the one hand we evoke the probabilistic modeling of SSCs and on the other the assumptions related to human reliability. From my point of view, this type of ambiguity can be avoided with the notion of "provision" which puts all the components of what I call the "safety architecture of the installation"			X	There is no ambiguity. Probabilistic modelling of SSC and of human actions have different methods and they have to be treated separated due to its intrinsic nature. The text meant to highlight the level of detail to be

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				fragility curves (if applicable) and reliability of digital instrumentation and control systems, including computer based systems used to control the process in the installation).	on the same level the the systems (material provisions) as well as the procedures (immaterial provisions).				achieved in the model following the application of the graded approach.
61.	Iran, Islamic Republic of	7	11/3.23Generalcomment	**~~Error! Reference source not found..~~**	This document needs to bereviewed by a technical editor.There are numerous syntax, and punctuation errorsthroughout.	X			The document was revised by the technical editors before posted. The Reference source error appeared after the conversion to .pdf file. In the revised version is corrected.
62.	Ukraine	7	para.3.23, 5.5, 5.6, 5.10, 6.1, 6.14, 7.3 and other	Broken references should be corrected	Editorial	X			The Reference source error appeared after the conversion to .pdf file. In the revised version is corrected.
63.	Ukraine	9	3.5	Incorrect references to ~~3.7~~3.6–3.7	Editorial	X			
64.	Sweden	5	3.7	"The ultimate product of a Level 2 PSA will be a description of the release categories with their related frequencies. The description …"	Editorial		X The ultimate product of a Level 2 PSA will be a description of a number of challenges to the containment, a description of the possible responses of that containment and an assessment of the consequent releases considering the source term calculations described by the release categories definitions, frequency and characterization of their magnitude.		To consider all important insights resulting from Level 2 PSA.
65.	Sweden	6	3.10	…qualification of personnel…	Editorial	X			

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66.	Sweden	7	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, and (iii) knowledge of PSA in general, and of Level 2 PSA techniques in particular. The depth of the team's expertise can be different depending on the stage in the lifetime of the plant at which the Level 2 PSA is carried out, the scope of the Level 2 PSA and the intended applications of the Level 2 PSA, but to the extent possible, extensive participation of the plant engineers and utility personnel, or designers if performed at the design stage, and probabilistic safety analysts specialized in accident phenomena and other Level 2 PSA disciplines is essential.	Example of very long sentences. 3.18 is only two sentences. There are many examples. Too long sentences makes it difficult to read and understand.		X The depth of the team's expertise can be different depending on the stage in the lifetime of the plant at which the Level 2 PSA is carried out, the scope of the Level 2 PSA and the intended applications of the Level 2 PSA.~~, but t~~To the extent possible, extensive participation of the plant engineers and utility personnel, or designers (e.g. if performed at the design stage), and probabilistic safety analysts specialized in accident phenomena and other Level 2 PSA disciplines is essential.		Text modified for better reading.
67.	Sweden	8	3.19	Recommend to place the paragraph in project management sub section in section 3.	Editorial			X	The paragraph is aimed at providing recommendations related to communication among team members, even though it is implemented by the project management.
68.	Sweden	9	3.21	"Experts in …"	I.e. remove "If possible", not needed. It is always up to the project to decide what experts are needed and how qualified they have to be in various expert areas. 3.21 also starts with stating "… team should consider including:"	X			
69.	Sweden	10	3.23	Problem with automatic referencing.	Editorial	X			
70.	Sweden	11	3.26	", and based on …"	Editorial			X	It refers to the methods and approaches.

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71.	Sweden	12	3.28 (f)	The probability development (e.g. data based and judgement based, phenomena probabilities);	Strange sentence. Maybe change to “The probability development (e.g. phenomena probabilities based on data or expert judgement);	X			
72.	WNA	16	4.1	The aim should be to identify and highlight plant SSCs and operating procedures that can influence the progression of severe accidents,	i.e., what I call the "provisions" of the safety architecture.			X	Agree the SSCs and the relevant operating procedures could be grouped as in the term “safety provisions”, however here is important to explicitly mention each of them.
73.	WNA	17	4.2 (b)	The flow paths from the area under the reactor pressure vessel to the main containment volume. Restrictions to the flow or other geometric aspects of the flow path will reduce the extent to which core debris is dispersed following a lower head failure. This is particularly important for high pressure melt ejection in a light water reactor;	In an optics of harmonization, it would be interesting to generalize the statement in order to cover also "non-conventional" technologies (e.g. MSR) or technologies which do not need pressure vessels (e.g. SFR or LFR). The case of LWR can be maintained as an example.			X	Last sentence of para 4.2 specify that these are examples of features for light water reactors. In addition, given the current knowledge available on licensed reactors technologies other that water cooled, examples, where consensus will be achieved, are difficult to present.
74.	Russian Federation	6	Para 4.3,Table 1	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 of a stationary fuel load" in the field «Comment» of Table 1 for the parameter "Radioactive material inventory".			X	This is not relevant for NPPs with online refueling.

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75.	Russian Federation	7	Para 4.3,Table 1	Accumulator volume and pressure set point and number (for each type of accumulators)	Clarification. It is proposed to clarify the parameter from Table 1 "Accumulator volume and pressure set point ".	X			
76.	Russian Federation	8	Para 4.3,Table 1	Containment design untightness/leakage and conditions of untightness/leakage	It is proposed to add the parameter "Containment design untightness/leakage and conditions of untightness/leakage" to the composition of the parameters from Table 1 for the container meter		X Added as: "Containment design leakage and conditions of leakage" And as comment "Actual operational values"		Leakage term is preferred rather than untightness.
77.	Russian Federation	9		No	For the parameter "Concrete aggregate" from table 1, it is necessary to clarify which concrete component is in question, because the composition of concretes of different components can differ significantly.		X Concrete aggregate of each containment structures		Modified to consider the different concrete used for different containment structure.
78.	Russian Federation	10	Para 4.3,Table 1	In-containment refueling water storage tank or refueling water storage tank or other in-containment water storage tank	Clarification. It is proposed to clarify the parameter "In-containment refueling water storage tank or refueling water storage tank" from Table 1.	X			

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79.	Ukraine	2	Para 4.3,Table 1	SPENT FUEL POOL (SFP)SFP geometry (shape, separation into sections, coolant inventory)Capacity and arrangement (number of stored spent fuel assemblies, racks design, loading pattern (if any))Decay heat (total decay heat at normal storage conditions and for emergency unloaded core)Radioactive material inventory (full inventory of radionuclides in SFP)Design parameters (coolant temperature and level)SFP safety features (flow rate, coolant inventory, soluble absorber concentration, temperature)SFP materials (steel, concrete, other)	Table 1 provides examples of key plant and/or containment design features which should be considered with respect of the progression and mitigation of severe accidents. Mentioned features include reactor, core, reactor coolant system and containment. SFP is one of the potential sources and/or contributors to the severe accident progression in the containment and needs to be considered	X			
80.	WNA	18		CONSIDERATIONS REGARDING MULTIPLE UNITS OR MULTIPLE RADIOACTIVE INSTALLATIONS ON A SITE	All the statements 4.4 to 4.9 are compatible with the notion of "safety architecture".			X	There is no recommendation. In addition, the presented terminology does not cover safety architecture since other accepted terms are already used.

81.	France	5	4.11	… to avoid large releases of radioactive substances to the environment. **In addition, the proper functioning of filtered venting systems in auxiliary building and leak of liquid effluent from reactor containment should also be considered.**	They could contribute to large releases of radioactive substances to the environment		X 4.11. For the plant familiarisation, the analyst should collect available documentation on the strategies implemented at the plant and become familiar with the priorities and actions contained within these strategies. Strategies developed to cope with severe accident progression generally include those aimed at (1) protecting the confinement function, including preventing the containment bypass, (2) if applicable, protecting the reactor building where the spent fuel pool is located. Depending on the reactor design, strategies may also address protection of the proper functioning of filtered venting systems in auxiliary building and management of leakage of liquid effluent from reactor containment in case of recirculation of contaminated water outside the containment. During the progression of a severe accident of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors), two important strategies are considered, firstly, in-vessel cooling and retention of damaged fuel (e.g. in-vessel melt retention for some reactor technologies such as water cooled, metal cooled and molten salt) and, secondly, ex-vessel cooling and retention of damaged fuel (e.g. ex-vessel corium cooling		Para modified to consider all relevant comments, including the comment proposed on the filtered venting system and the liquid effluents.

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							for some water cooled reactor designs). See also paras 4.14–4.15.		

82.	Iran, Islamic Republic of	1	4.11 page 16	For achieving the fundamental safety objective, strategies to cope with severe accident progression should be defined to preserve the integrity of the reactor containment and, if applicable, of the reactor building where the spent fuel pool is located and preventing containment bypass.	In addition to preserve the integrity of the reactor containment, consideration shall be given to prevent containment bypass (for example by leakage form primary to secondary circuit).strategies for maintaining containment integrity and preventing bypass are of the highest priority once the mitigatory domain is entered. The concept of containment bypass can not be included in the loss of containment integrity. Because bypass mainly happens through the pipes connected to the primary circuit while the integrity of the containment is maintained.		X 4.11. For the plant familiarisation, the analyst should collect available documentation on the strategies implemented at the plant and become familiar with the priorities and actions contained within these strategies. Strategies developed to cope with severe accident progression generally include those aimed at (1) protecting the confinement function, including preventing the containment bypass, (2) if applicable, protecting the reactor building where the spent fuel pool is located. Depending on the reactor design, strategies may also address protection of the proper functioning of filtered venting systems in auxiliary building and management of leakage of liquid effluent from reactor containment in case of recirculation of contaminated water outside the containment. During the progression of a severe accident of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors), two important strategies are considered, firstly, in-vessel cooling and retention of damaged fuel (e.g. in-vessel melt retention for some reactor technologies such as water cooled, metal cooled and molten salt) and, secondly, ex-vessel cooling and retention of damaged fuel (e.g. ex-vessel corium cooling		Reference to the confinement function added which covers both the protection of the containment integrity and the prevention of bypasses.

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							for some water cooled reactor designs). See also paras 4.14–4.15.		

83.	WNA	19	4.11	During the progression of a severe accident in the reactor core, two main strategies are considered for the damaged fuel, depending on the reactor design and technology: in-vessel melt retention and ex-vessel corium cooling.	This sort of statement should be formulated to generalize and to address alternative technologies.		X 4.11. For the plant familiarisation, the analyst should collect available documentation on the strategies implemented at the plant and become familiar with the priorities and actions contained within these strategies. Strategies developed to cope with severe accident progression generally include those aimed at (1) protecting the confinement function, including preventing the containment bypass, (2) if applicable, protecting the reactor building where the spent fuel pool is located. Depending on the reactor design, strategies may also address protection of the proper functioning of filtered venting systems in auxiliary building and management of leakage of liquid effluent from reactor containment in case of recirculation of contaminated water outside the containment. During the progression of a severe accident of the fuel in the reactor vessel (e.g. in the reactor core for water cooled reactors), two important strategies are considered, firstly, in-vessel cooling and retention of damaged fuel (e.g. in-vessel melt retention for some reactor technologies such as water cooled, metal cooled and molten salt) and, secondly, ex-vessel cooling and retention of damaged fuel (e.g. ex-vessel corium cooling		Proposed text to be more technology inclusive.

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							for some water cooled reactor designs). See also paras 4.14–4.15.		
84.	Sweden	13	4.12	" …availability of electricity, compressed air or water sources."	Editorial	X			
85.	Egypt	4	4.13	Paragraphs 4.14 - 4.15 provide recommendations on relevant information on safety provisions	Paragraphs discuss provisions that should be collected in the familiarization task start from: 4.14 to 4.15.	X			
86.	Sweden	14	4.13	Strange referencing "0-4.15"	Editorial	X			
87.	Japan	3	4.14	For water cooled reactors, the in-vessel melt retention strategy is aimed at ensuring a passive and/or active reflooding of the reactor pressure vessel cavity up to a level to ensure and maintain, with sufficient confidence, the integrity of the reactor pressure vessel by cooling it from outside and the integrity of the corium inside by in-vessel water.	To unify the terminology (see 4.14(a)).	X			
88.	Japan	4	4.14(f)	Water inventory available (i.e. affecting the delay the time of corium arrival in the lower plenum and therefore reduce the heat amount of corium ~~residual power to extract~~).	Main function of the water in the lower plenum is to reduce heat amount of corium.			X	The heat produced by the corium comes from the residual power generated by the mix of fuel in the corium itself. Therefore, it is residual power.

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89.	WNA	20	4.16	Requirement 19 of GSR Part 4 (Rev.1) **Error! Reference source not found.** states that "**Data on operational safety performance shall be collected and assessed.**" When the PSA team has developed a general understanding of the plant design and features that may influence severe accidents and releases of radioactive material, the quantitative data that are necessary to carry out the plant specific analysis should be collected and organized. The data necessary for the PSA depend in part on the scope of the analyses and the nature of the computational tools. For example, the amount and type of input data collected may depend on the plant specific computer model used to calculate accident progression. Detailed architectural and construction data for the containment structure should be collected to develop plant specific model calculations of the containment performance if such calculations are required by the scope of the containment performance analysis.	This sort of recommendation could make explicit reference to the achievement of a PIRT analysis.		X Text modified as:Requirement 19 of GSR Part 4 (Rev.1) [2] states that "Data on operational safety performance shall be collected and assessed." When the PSA team has developed a general understanding of the plant design, phenomena[12] and features that may influence severe accidents and releases of radioactive material, the quantitative data that are necessary to carry out the plant specific analysis should be collected and organized.And Footnote 12 as: Source of information for the phenomena could be obtained from the Phenomena Identification and Ranking Table (PIRT) analysis for severe accidents, if available.		

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90.	Germany	8	4.17	Data should be obtained from qualified sources, such as: (a) Design documents and/or plant licensing documents~~;~~, such as safety analysis report , technical specifications for the plant, system(s) descriptions; (b) As built drawings; (c) Plant specific normal operating, maintenance or test procedures; (d) Main automatisms, emergency operating procedures and severe accident management guidelines; (e) Engineering calculations or analysis reports; (f) ~~Observations during plant walkdowns;~~ Plant walkdown reports;(g) ~~Construction standards~~ Regulatory requirements; (h) Vendor manuals. (i) Other relevant plant documents.References to the source(s) of data should be recorded as part of the PSA documentation.	We suggest to be consistent with DS523, as that list is more comprehensive.		X (a) Design documents and/or plant licensing documents, such as safety analysis report, technical specifications, system(s) descriptions; (b) As built drawings; (c) Plant specific normal operating, maintenance or test procedures; (d) Information on plant automatic actuations; (e) Emergency operating procedures and severe accident management guidelines; (f) Engineering calculations or analysis reports; (g) Observations during plant walkdown reports and/or walkdown reports; (h) Construction standards; (i) Regulatory requirements; (j) Vendor manuals; (k) Other relevant plant documents.		Observations during plant walkdowns as well as construction standards are applicable since they provide information on materials used by SSCs, which impact severe accident phenomena.
91.	France	13	5.5	Examples of such attributes for water cooled reactors are given in ~~**Error! Reference source not found**~~ **Table 3**.	Correction of an incorrect link to a reference.	X			
92.	Sweden	15	5.5	Problem with automatic referencing.	Editorial	X			
93.	Russian Federation	11	Para 5.5,Table 3	Containment passive heat removal system (available/unavailable	It is proposed to add to Table 3 the following attribute for status of containment's engineered safety features: «Containment passive heat removal system (available/unavailable»)		X Text added as:Containment passive heat removal system (if any):— Available— Unavailable— In operation— Failed		Availability and unavailability do not consider the if the system is in operation or failed.
94.	ENISS	4	5.6	"If the Level 2 PSA is developed following a separated approach (see para 2.52.6) Level 1 PSA […]"	Reference to Para. 2.6 seems more adequate.	X			

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95.	France	14	5.6	In such cases, the Level 1 PSA should be expanded to take into account the missing aspects in the specification of PDSs (see ~~**Error! Reference source not found**~~. for reference).	Error in the reference link (not sure of the reference to which it is linked)	X			
96.	Sweden	16	5.6	Problem with automatic referencing.	Editorial	X			
97.	Sweden	17	5.6	"..in FIG. 1 in section 1, thereby…"	Editorial			X	There is only one FIG 1 in the draft.
98.	ENISS	4	5.7	"If the Level 2 PSA is developed as part of an integrated Level 1 – Level 2 PSA (see para ~~2.5~~**2.6**) the Level 1 PSA integrates the containment systems."	Reference to Para. 2.6 seems more adequate.	X			
99.	ENISS	4	5.9	"It should be noted that the level of detail of characteristics used to define the PDSs depends on the case used for the development of Level 1 PSA and Level 2 PSA (see para ~~2.5~~**2.6**)."	Reference to Para. 2.6 seems more adequate.	X			
100.	ENISS	5	5.9	"[…] If the 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. If the Level 1 PSA and the Level 2 PSA are an integrated model developed in a linked event tree or linked fault tree software many of characteristics listed later in paras. 5.10-5.12 will be implicitly available for the Level 2 PSA to use without being made explicit for the PDSs definition. Such an approach may allow to reduce the number of PDSs needed**.** ~~In other words~~ In any case, even though the structure of the PDSs would be simpler in an integrated Level 1 PSA and Level 2 PSA model, the analyst should verify that simplifications or assumptions in Level 1	Proposal to emphasise the implications.	X			

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				PSA model will not screen out possible PDSs contributing to radioactive releases."					
101.	France	15	5.10	Depending on the reactor technology, examples are the following (see ~~**Error! Reference source not found.**~~ for further details for water cooled reactors):	Error in the reference link (not sure of the reference to which it is linked)	X			
102.	Japan	5	5.10(j)	The state of the suppression pool (e.g. subcooled or saturated) when core damage occurs (e.g., for a boiling water reactor).	The pool which should be considered in severe accidents with core damage may be a suppression pool.Expression consistence with (i), because both mention about the pool with pressure suppression capability.	X			
103.	Sweden	18	5.10	Problem with automatic referencing.	Editorial	X			
104.	France	16	5.13	The first is to combine similar PDSs and perform a bounding analysis to select a representative sequence that characterizes the PDS for the purpose of the Level 2 PSA. **For instance, if the Level 2 PSA relies on time consuming physical calculations, it could be possible to run a manageable number of these calculations and attribute the outcomes of one calculation to several PDSs which are similar in regard of the accident progression. This could allow to deal with a large amount of PDSs without running a non-manageable number of physical calculations.**	Proposition of an example of what is done in IRSN to illustrate that it is possible to deal with a large amount of PDSs without running too much physical calculations which can be very time consuming.	X			
105.	Sweden	19	5.13	"…a significant underprediction of the risk…"	Editorial, not "under prediction". Maybe should be phrased "underestimation"?	X			

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106.	Germany	9	5.16	In order to extend the scope of Level 2 PSA to include internal and external hazards such as fire, seismic ~~events~~ hazards and external flooding, the impact of ….	Precision for consistency with other SSGs and TECDOCS on external events and hazards	X			
107.	ENISS	6	5.17 and footnote 10	~~Dependent failures[10]~~	To clarify / Isolation function is part of the confinement function		X 5.17. In addition to para 5.16, the potential impact of hazards on the systems ensuring the confinement function ~~as well as the dependent failures which can be induced by the hazards~~ should be taken into account as part of the Level 2 PSA, if those aspects have not yet been taken into account in the Level 1 PSA output. Footnote added after confinement function as: Typical examples of impacts from hazards are failures of the isolation function of systems ensuring the confinement function due to internal fire, explosion or flooding at the plant, damage of the containment due to seismic events, aircraft crashes or external explosions (blasts).		Para 5.16 already mentions systems necessary for mitigation of severe accidents, including systems that support operator actions, and the impact on the integrity of the containment. The mention of confinement function in 5.17 complements para 5.16.

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108.	ENISS	7	5.19	"The analysis process to be conducted for considering hazards and their combinations for Level 1 PSA is described in paras 6.4-6.27 of SSG-3 (Rev. 1) [4]. This process is applicable to Level 2 PSA and it is not repeated here. For the Level 2 PSA, single as well as combined hazards have the potential to result in accident sequences induced by common cause initiators, such as~~Some examples are: — A~~ a design extension condition earthquake resulting in a station blackout and a containment failure, perhaps with consequential internal fire or flooding; — ~~A flooding and high winds combined event that might lead to the loss of the heat sink together with the loss of off-site power; — An aircraft crash causing a common loss of offsite power and emergency diesel generator failure, which again does not only result in a plant transient but an accident sequence with containment bypass and releases of radionuclides (airborne or via water path).~~"	It appears necessary to develop a Level 2 PSA for initiating events that impact the containment directly (example 1). Since the second and third examples do not affect directly containment, the need to extend this analysis to Level 2 PSA seems not automatic.		X For the Level 2 PSA, single as well as combined hazards have the potential to result in accident sequences induced by common cause initiators that might impact the confinement function.		All the examples of combined hazards presented might affect the confinement function for which Level 2 PSA might need to be developed.
109.	Egypt	5	5.19	The analysis process to be conducted for considering hazards and their combinations for Level 1 PSA is described in paras 6.1 - 6.25	Paragraphs considering hazards and their combinations for Level 1 PSA start from: 6.1 - 6.25 in SSG-3.		X 5.19. The analysis process to be conducted for considering hazards and their combinations for Level 1 PSA is described in paras 6.1-6.27 of SSG-3 (Rev. 1) [4].		Updated with the version of SSG-3 provided for preprint.
110.	Germany	10	5,19 second bullet	A combination of external flooding and high winds ~~combined event~~ hazards …	Precision for consistency with other SSGs and TECDOCS on external events and hazards	X			

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N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
111.	ENISS	8	5.20	"In order to be widely applicable, the Level 2 PSA for hazards should be based on a full scope Level 1 PSA covering hazards as described in SSG-3 (Rev. 1) [4]. This requires that the Level 1 PSA: (a) Does not only include a comprehensive set of internal initiating events, but also a set of relevant internal and (natural and human induced) external hazards including combined hazards as defined in SSG-64 [6] and SSG-3 (Rev. 1) [4]; (b) Covers all relevant plant operational states, which may include ~~including~~ start-up, operation at full power and low power, and all modes occurring during plant shutdown and refuelling. This will ensure that the insights from the PSA relating to the risk significance of accident sequences, SSCs, human errors, common cause failures, etc. are derived from a comprehensive, integrated model of the plant. It should be noted that the development of a Level 2 PSA for hazards depends on the scope set but can also be influenced by the L1 hazards PSA results. In particular in case of a low strength of knowledge associated to the Level 1 PSA results, the relevance of extending this PSA to Level 2 should be analyzed with regards to safety issues, feasibility and ease of analyzing insights from it."	In some cases, for example when hazards are foreseeable (such as flooding from a river) it can be considered that the plant is not anymore in operation at full power at the time of the flooding. The development of a Level 2 PSA should be focused on relevant hazards. If there is too much uncertainty / too limited strength of knowledge in the L1 PSA results, the need to develop a Level 2 PSA should be questioned.		X "In order to be widely applicable, the Level 2 PSA for hazards should be based on a full scope Level 1 PSA covering hazards as described in SSG-3 (Rev. 1) [4]. This requires that the Level 1 PSA: (a) Does not only include a comprehensive set of internal initiating events, but also a set of relevant internal and (natural and human induced) external hazards including combined hazards as defined in SSG-64 [6] and SSG-3 (Rev. 1) [4]; (b) Covers all plant operational states. This will ensure that the insights from the PSA relating to the risk significance of accident sequences, SSCs, human errors,…		GSR, Part 4 requires that hazards are analysed for all plant operational states. This is also required in SSG-3, Section 6. In principle SSG-3 and SSG-4 require analysing hazards for all POS in line with GSR, Part 4. Non-full scope studies should not be explicitly addressed; limited scope considerations are provided in 5.21; in addition.

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112.	ENISS	9	5.22	~~"Those hazards, single as well as combined ones, which were screened out from further analysis within the Level 1 PSA should also be reassessed to consider if such hazards should be taken into account in Level 2 PSA. In this context, it should be distinguished between: (a) Hazards, for which the site and plant specific screening has demonstrated that they do not need to be analysed in detail but that a rough probabilistic estimate of the Level 1 PSA PDSs (core and/or fuel damage) is sufficient, detailed accident sequences do not have to be modelled, but again rough estimates of the radioactive release frequencies (large release frequency or large early release frequency) are sufficient; (b) Hazards, for which detailed accident sequences have to be modelled and quantified within Level 2 PSA.~~The potential for hazards or combined hazards to affect the containment should be addressed during the screening process. If those hazards were screened out with criteria based on Level 1 PSA only, they should be reassessed considering Level 2 PSA issues."	Proposed rewording to focus only on the need to take account in the screening process the hazards or combined hazards that may affect the containment in the context of L2 PSA. Moreover, the screening process is supposed to be a simple analysis and not a dedicated PSA model as it might be understood in current the para. 5.22.		X 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 SSG-3, Rev.1 [4] paras. 6.17 to 6.19, noting that the latter explicitly states that "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." To determine if such hazards should be taken into account in Level 2 PSA, it should be considered if they can affect the confinement function. In this context, it should be distinguished between: (a) Hazards, for which the site and plant specific screening has demonstrated that they do not need to be analyzed in detail, but that a bounding assessment of the Level 1 PSA PDSs (core and/or fuel damage) is sufficient, detailed accident sequences do not have to be modelled, but again a bounding assessment of the radioactive release frequencies (large release frequency or large early release frequency) is sufficient; (b) Hazards, for which detailed accident sequences have to be modelled and quantified within Level 2 PSA.		SSG-3 (e.g. 6.17 ff., Fig. 2) presents the hazards screening for single and combined hazards, 6.18 the qualitative screening criteria, 6.19 general guidance on quantitative screening criteria without directly prescribing reference values (thresholds) and requires "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." The original text has been improved for more clarity including a precise reference to the corresponding paras of SSG-3, including a precise reference to the corresponding paras of SSG-3, and – also in line with SSG-3 - a changed terminology ("bounding assessment").

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113.	Germany	11	5.22 (a)	Hazards, for which the site and plant specific screening has demonstrated that they do not need to be analysed in detail but that a ~~rough probabilistic estimate~~ bounding assessment of the Level 1 PSA PDSs (core and/or fuel damage) is sufficient, detailed accident sequences do not have to be modelled, but again ~~rough estimates~~ a bounding assessment of the radioactive release frequencies (large release frequency or large early release frequency) ~~are~~ is sufficient;	Correction in line with DS523, more precision		X 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 SSG-3, Rev.1 [4] paras. 6.17 to 6.19, noting that the latter explicitly states that "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." To determine if such hazards should be taken into account in Level 2 PSA, it should be considered if they can affect the confinement function. In this context, it should be distinguished between: (a) Hazards, for which the site and plant specific screening has demonstrated that they do not need to be analyzed in detail, but that a bounding assessment of the Level 1 PSA PDSs (core and/or fuel damage) is sufficient, detailed accident sequences do not have to be modelled, but again a bounding assessment of the radioactive release frequencies (large release frequency or large early release frequency) is sufficient; (b) Hazards, for which detailed accident sequences have to be modelled and quantified within Level 2 PSA.		SSG-3 (e.g. 6.17 ff., Fig. 2) presents the hazards screening for single and combined hazards, 6.18 the qualitative screening criteria, 6.19 general guidance on quantitative screening criteria without directly prescribing reference values (thresholds) and requires "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." The original text has been improved for more clarity including a precise reference to the corresponding paras of SSG-3, including a precise reference to the corresponding paras of SSG-3, and – also in line with SSG-3 - a changed terminology ("bounding assessment").

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114.	WNA	21	5.22	Those hazards, single as well as combined ones, which were screened out from further analysis within the Level 1 PSA should also be reassessed to consider if such hazards should be taken into account in Level 2 PSA.	I understand the text, but I don't understand the logic. If a hazard has not been considered for the PSA level 1, this means that we do not know the "environmental conditions" generated by the hazard in question and which could characterize the PDS to be taken into account for the PSA level. 2 (?). Is it possible to clarify the statement?		X 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 SSG-3, Rev.1 [4] paras. 6.17 to 6.19, noting that the latter explicitly states that "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." To determine if such hazards should be taken into account in Level 2 PSA, it should be considered if they can affect the confinement function. In this context, it should be distinguished between: (a) Hazards, for which the site and plant specific screening has demonstrated that they do not need to be analyzed in detail, but that a bounding assessment of the Level 1 PSA PDSs (core and/or fuel damage) is sufficient, detailed accident sequences do not have to be modelled, but again a bounding assessment of the radioactive release frequencies (large release frequency or large early release frequency) is sufficient; (b) Hazards, for which detailed accident sequences have to be modelled and quantified within Level 2 PSA.		SSG-3 (e.g. 6.17 ff., Fig. 2) presents the hazards screening for single and combined hazards, 6.18 the qualitative screening criteria, 6.19 general guidance on quantitative screening criteria without directly prescribing reference values (thresholds) and requires "Hazards of very low frequency but with potentially severe consequences in terms of releases of radioactive material should be considered for the purposes of a Level 2 PSA." The original text has been improved for more clarity including a precise reference to the corresponding paras of SSG-3, including a precise reference to the corresponding paras of SSG-3, and – also in line with SSG-3 - a changed terminology ("bounding assessment").

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115.	France	6	6.1	Section **Error! Reference source not found**.	Error	X			
116.	Sweden	20	6.1	Problem with automatic referencing and strange reference to section 0.	Editorial	X			
117.	Sweden	21	6.5	Remove?	Is this type of guidance needed to be repeated throughout the same guide? Should be enough in the beginning, in section 2.			X	It is particular important to recall in this section since it is related to the performance of severe accident progression simulation.

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118.	Russian Federation	12	6.8	Each identified PDS should be mapped to specific representative calculations, however some calculations can represent more than one PDS, if justified without significant conservative assumptions. In addition, calculations could also be performed for those PDSs that may have a low occurrence frequency, but which have the potential to result in large and/or early releases of radionuclides to the environment. Such PDSs typically involve either direct containment bypass or early failure of the primary and/or secondary containment.	Proposed:1) In the first sentence of Item 6.8, after the word "Each", add "identified", and delete the phrase "representing a significant contributor to core damage", since at process of PDSs identification, their possible grouping was made and/or the number of PDS is limited by the accepted value of the PDS frequency (see Item 5.13). Therefore, all identified PDS should be considered as significant. If the specified phrase is not deleted, it should be determined what is it means "significant contributor to core damage". For example, as a significant PDS, PDS can be considered, the frequency of which is more than 1% of the established probabilistic safety criterion for LRF and/or LERF.2) To avoid excessive conservativeness, add "without significant conservative assumptions" at the end of the first sentence.3) To exclude a last sentence, because a) In the first sentence of paragraph 6.8, it was already said about the use of calculation analyzes for different PDS; b) Calculations must be performed for all identified PDS - see comment 1) above.		X Comment 1 and 2 Text modified as:6.8. Each identified PDS representing a significant contributor to core damage[17] (see para 5.13 and footnote 16) should be mapped to specific representative calculations, however some calculations can represent more than one PDS, if justified. Footnote added to para 5.13 as: In some Member States a cut-off value in terms of percentage of the total risk metric (Large Release Frequency or Large Early Release Frequency) is established to consider significant PDSs from less important PDSs.	X	The mention related to "representing a significant contributor to core damage" is maintained and a footnote is added in para 5.13. There is no need to specify "without significant conservative assumptions" since the purpose of PSA is to be realistic as possible. Comment 3: The information here provided doe not intent to avoid calculations but to acknowledge the amount of information that will be generated.
119.	Japan	6	6.9	Para 6.9 should be deleted.~~If relevant, Level 2 PSA should also consider assessment of reactivity accident scenarios resulting in prompt criticality accidents leading to reactor core damage and potential damage to the containment integrity.~~	The guide of Level 1 PSA (DS523, revision of SSG-3) has already taken the reactivity accident into account as initiating event. Thus, the description regarding the reactivity accident should not be specified in the guide of Level 2 PSA (this DS528, revision of SSG-4).			X	If the reactivity accident leads to containment failure, it needs to be considered as part of the Level 2 PSA.

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120.	Ukraine	4	para 6.10 line 2;para.10.8 (a,b); para.11.15	Incorrect references to Annex II should be replaced with references to Annex I	Editorial	X			All, 6.10, 10.8 and 11.15 were corrected.
121.	Sweden	22	6.12	Almost the same as 6.5	See comment also on 6.5			X	Agree that they complement each other, while 6.5 insists on the training, 6.12 insist on the knowledge about the code.
122.	France	7	6.14	they end ~~with the release of radionuclides into environment~~ **when the most part of the release of radionuclides into environment has been released, or after corium stabilization (in-vessel or ex-vessel)**.	Integral analyses must not stop with the first releases.		X …Integral analyses start with the initiating event and end according to appropriate criteria, depending on the purpose of the analysis. Examples of criteria for termination of analyses that have been used are 1) when the cumulative release of radionuclides into the environment has stabilised, 2) after corium stabilization (in-vessel or ex-vessel), or 3) after a pre-determined mission lime has elapsed. …		Text modified to cover all potential possibilities.
123.	France	8	6.14	Section **Error! Reference source not found.**	Error	X			
124.	Sweden	23	6.14	Problem with automatic referencing	Editorial	X			

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125.	Japan	7	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 the design of nuclear power plants, are not useful or productive in severe accident analyses for Level 2 PSA___because the conservative assumptions may lead to deviation from optimal severe accident management strategies and severe accident analysis results.	The reason why conservative assumptions for the severe accident analyses are not useful or productive, should be explained.		X 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 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, and consequently may lead to deviations from optimal severe accident management strategies.		Modification of text for better reading
126.	France	9	6.17	… should be considered in the severe accident analyses. **Their moment of realization should be representative of reality.**	Ensuring representativeness of analyses.		X 6.17. Severe 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.		For better reading.
127.	Sweden	24	6.19	"…guidance…"	Guidance better word than recommendations	X			
128.	USA	9	6.22	Original wording: "Specific analysis should be performed for low power and shutdown modes of reactor operation." Reword "if a low power and shutdown level 2 PSA is pursued, specific analysis should be performed…"	Original wording implies that low power and shutdown PSA is required. Revised wording should be clearer, if a low power and shutdown is pursued, specific analysis should be performed.			X	Performing Level 2 PSA for LP&SD is of particular importance since most of the time containment building may be open or partially opened, therefore it is recommended.

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129.	Russian Federation	13	6.26	A plant specific list of uncertain parameters to be varied in the frame of the uncertainty/sensitivity analysis should be derived.At forming the list of parameters for uncertainty analysis, it should not include as parameters a correlation coefficients, model parameters, etc. used in modeling the phenomenology of severe accidents in the corresponding computer codes, established as part of the computer code validation procedure. Otherwise, their variation can lead to completely incorrect results of the uncertainty analysis.	In order to avoid obtaining incorrect results of uncertainty analysis when varying parameters.			X	There is no need to specify those parameters since this is a matter having qualified analysts and follow code developer recommendations as regard with modifying parameter values that are built into the code. In addition, some of the listed parameters may need to be explored as art of the uncertainty analyses.
130.	Japan	8	7.2(a)	The capability of the containment to maintain its leak tightness under internal ~~pressurization~~ loads (para 7.4-7.11);	To unify the terminology (see the title before para. 7.4).	X			
131.	Sweden	25	7.2	…leaktightness…	Editorial	X			
132.	Sweden	26	7.3	Problem with automatic referencing	Editorial	X			
133.	France	10	Table 4	Additional type of severe accident event: **Radioactive releases into the environment**Related phenomena: **Containment break size Containment leak rateReleased fraction of inventoryIodine chemistry**	Lot of uncertainties associated to assessment of releases.		X added in Table 9		Table 9 identifies issues related to uncertainties for source term calculations as the text in the comment proposes.
134.	Russian Federation	14	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			X	Unfortunately, that proposal was not considered during the drafting of the safety guide as the level of detail to be presented in the safety guide.

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					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.				
135.	France	11	7.4	known as a fragility curve **or a fragility (hyper)surface**.	Not necessary a curve when both pressure and temperature are retained		X2 Footnotes added for clarification as: Fragility curve representing the probability of containment failure as a function of one variable, such as pressure or temperature. Fragility surface representing the probability of containment failure as a function of more than one variable together, such as pressure and temperature.		For clarification.
136.	Sweden	27	7.10	Problem with automatic referencing	Editorial	X			
137.	ENISS	10	7.12	Delete paragraph 7.12+ proposal to modify the subtitle which precedes: "Analysis of containment leaktightness due ~~to failure mechanisms induced by severe accident phenomena~~ molten core–concrete interactions"	The introduction of physical phenomena generating pressure loading on containment (hydrogen combustion) seems redundant with the previous paragraphs: paras 7.4 to 7.11 already deal with this risk. The objective of this sub-section (paras 7.13 to 7.16) seems to be to focus on MCCI which is a mechanism that may threaten containment integrity differently than pressure loading.		Title changed as "Analysis of containment leaktightness due to other failure mechanisms induced by severe accident phenomena". Para 7.12 modified as:7.12. Containment leaktightness might be also affected by failure mechanisms induced by severe accident phenomena. Examples of phenomena to consider could be induced fires (e.g. graphite fires), steam explosion (e.g. instantaneous vaporization of water induced by its contact with molten corium), chemical attack (e.g. chemical reactions affecting containment structures integrity) and direct contact between molten core	X	The list of mechanisms induced by severe accident phenomena is larger than the combustion and MCCI. Para modified to provide further examples.

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							debris and containment structures. Recommendations related to the consequences of molten core debris and containment structures for the containment integrity analysis are presented in paras 7.13 to 7.16.		
138.	Sweden	28	7.15	Almost the same as 7.13. Consider merging.	Editorial			X	The initial subject of the recommendation is the same, but the examples are different.
139.	France	4	7.17	The potential for containment isolation failure should be assessed. All the containment penetrations should be **modelled or a** carefu~~lly~~ **justification has to be provided to justify the screen out of some penetration** ~~analysed to decide if they should be modelled or not~~. **In a preliminary version of a level 2 PSA**, screening criteria may be applied in order to focus on the relevant penetrations that are most likely to result in important releases. 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) or for closed loop systems inside the containment provided that closed loop integrity will not be threatened during the accident.**If any, the plant operating feedback regarding containment isolation valves leakages shall be taken into account**.	In a mature level 2 PSA, even the penetration normally closed during accident could be considered to identify the most risk significant pre-accidental error (wrong position) or to take into account plant operating feedback regarding containment isolation valves (CIV) leakages. In addition, the screen out of permanently closed CIV is not consistent with 7.18Regarding the containment penetration connected to closed loop system inside the containment, the demonstration that these systems are always robust to severe accident conditions (temperature, pressure, structure displacement due to load (hydrogen combustion, steam explosion, DCH…)) seems much more difficult to reach than modelling the penetration.Several utilities perform dedicated plant operating feedback for CIV. These inputs have to be valorised in L2PSA.		X …~~In a preliminary version of a level 2 PSA,~~ Screening criteria may be applied in…		There is no need to specify the preliminary version of Level 2 PSA since the screening is valid to all Level 2 PSA models and not only to preliminary models.
140.	Sweden	29	7.24	Material variability and modelling uncertainty can be …	Editorial	X			

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141.	Sweden	30	7.25	Strange reference to para 0	Editorial	X			
142.	Egypt	6	7.28	The molten core–concrete interactions (MCCI) phenomenology is rather complex and various situations may occur as the result of the accident progression. Assessment of the probability of an extensive erosion of structures should account for the uncertainties affecting the MCCI calculations.	The abbreviation should be defioned in the text when used at fiest time.		X 7.28. The molten core–concrete interactions phenomenology is rather complex and various situations may occur as the result of the accident progression. Assessment of the probability of an extensive erosion of structures should account for the uncertainties affecting the molten core–concrete interactions~~MCCI~~ calculations.		Abbreviation was replaced by the full text according to IAEA rules for IAEA publications.
143.	Sweden	32	8	Missing guidance on revisiting level 1 operator actions from a level 2 perspective.	Depending on the objectives etc. with PSA level 1, it is advised to revisit the PSA level 1 HRA to reassess level 1 operator actions from a PSA level 2 perspective. E.g. conservatism may have been used resulting in too high numbers.		X added as new para 8.4 as: 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).		
144.	Ukraine	3	para. 8.7 line 1	8.7 ~~Annex I~~ Appendix I provides more detailed information about performing human reliability analysis for a Level 2 PSA	Editorial. Annex I is related to computer codes for SA simulation. Correct reference for human reliability assessment is Appendix I	X			
145.	France	12	8.9	….dependencies between the human actions credited in Level 1 PSA and Level 2 PSA. **Especially if : these human actions are carried out by the same operatorssame equipments are requiredsevere accident occurs quickly…**	Detail the cases where dependencies must be taken into account.		X … 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.		Minor language modifications.
146.	Sweden	33	8.10	Delete "as part of the event tree logic"	It is not necessary to consider in the event tree logic. Other ways can be used. Therefore suggest to delete this part of the sentence.		X … actions should be considered (e.g. as part of the event tree logic).		The event tree logic presented as an example

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147.	Japan	9	8.13	Assessment of the reliability of equipment credited within the Level 2 PSA should consider the periodic testing and maintenance practices or planned procedures. Such practices or procedures may differ from those used for the systems and components credited within Level 1 PSA ~~to prevent core damage~~ and thus may have an influence on systems reliability.	To show the relationship between Levels 1 and 2 more clearly.	X			
148.	ENISS	11	8.14	"Adverse environmental impacts may include ~~high~~ containment/auxiliary buildings high temperature, pressure, humidity and radiation conditions. ~~Examples are energetic events (e.g. short term temperature and pressure spikes or impulse loadings from detonations or steam explosions) could affect equipment reliability (e.g. the electronic instrumentation, rubber gaskets that could be vulnerable to high radiation)"~~ 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 (e.g. electronic instrumentation, rubber gaskets could be vulnerable to high radiation)."	Proposed rewording for clarity	X			

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149.	France	2	8.15	…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**.	The question of the mission time considered to assess the probability of systems failure is not addressed. However, some equipment's (for example for long term residual heat removal) may be required for month or years and my not be repairable (due to the dose). This L2PSA specificity (compare to L1PSA) should be mentioned.	X			
150.	Ukraine	10	Table 6, item 20	The text in "Dependencies" column "4, 8, 9, 10б" shall be replaced with "4, 8, 9, 10"	Editorial	X			
151.	ENISS	12	Section 9Terminology Comment	Section 9 (paras. 9.16 and 9.17) and Section 10 use the following terminologies: "initial release category" (that is defined as L2 PSA "end states") and "final release category" (that represents some grouping of "initial release categories" use for source term calculations). Section 1 rather use the terminologies: "release category" and "source term category". Even if footnotes 1 and 16 explain that these terminologies are globally synonymous, it would be more comfortable for the reader to have clear definitions of these terms and consistent usage throughout the document.Note: Para 11.3 also use the terminology "release classes".				X	The terms "release categories" and "source term categories" are used in the references interchangeable as explained in footnotes 1 and 16.In para 11.3 and 11.6 "release classes" were modified to "release categories" for consistency.
152.	ENISS	4	9.1	"For the development of a Level 2 PSA event tree model, two different approaches can be used: an integrated	Reference to Para. 2.6 seems more adequate.	X			

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				approach and a separated approach which differ mainly by the way information is transmitted from Level 1 PSA to Level 2 PSA (see para ~~2.5~~**2.6**)."					
153.	Russian Federation	2	9.2	Although containment event trees have historically been used for Level 2 PSAs, accident progression event trees were introduced in NUREG-1150 [22] and adopted in the ASAMPSA2 project [21]. This term is used consistently throughout this Safety Guide (see para. 1.8(c)). In practice accident progression event trees involve a greater level of phenomenological modelling, whereas containment event trees are structured to focus on containment challenges and event tree top events (also referred to as nodal questions) with phenomenological processes and associated events included in the top event supporting logic. Both approaches applied consistently should result in equivalent Level 2 event tree end states.	The term "Containment Event Tree" is practically not used in this guide, therefore, it is proposed to exclude the mentioned text fragment from the main text of Item 9.2, and bring it as a footnote to Item 1.8(c). The proposed in comments 2 and 3 systematizes the information in the guide.		X Para 9.2 modified as: "In Level 2 PSAs, event trees are used to delineate the sequence of events and severe accident phenomena after the onset of core damage that challenge containment integrity and the successive barriers to radioactive material release. They provide a structured approach for the systematic evaluation of the capability of a plant to cope with severe accidents. Their use is shown in Fig. 1. Such event trees, termed accident progression event trees in this guide, include modelling of phenomena, systems actuation or failure, human actions and all impacts on the confinement of radioactive products or the radioactive releases in the environment.		See answer to comment 11 The term "containment event tree" is deleted.
154.	Russian Federation	3	9.3	Nodal questions of the containment event tree should also address issues and actions relating to severe accident management.	It is proposed to delete footnote 12 if the comment to Item 9.2 related to term "Containment Event Trees" will take into account.		X footnote modified as: Nodal questions also address issues and actions relating to severe accident management.		See answer to comment 11
155.	Sweden	34	9.3	Replace "material" with "SSCs"	Material seem to be the wrong word here.	X			
156.	Sweden	35	9.11	If possible, write out the reference "NUREG-1150 [22]"	Makes it easier to read and understand. The same comment may apply in more places.	X			

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157.	Sweden	36	9.11	"…since then…"	Editorial			X	The introduction of NUREG-1150, makes sentence readable.
158.	Egypt	7	9.12	Experimental programmes regarding the response of containments to internal pressurization design extension conditions that may be useful in supporting development of containment fragility models is provided in Ref. [19].	The term "beyond design basis conditions" is no longer used in IAEA publications.		X Text modified for clarification as: 9.12. Experimental programmes regarding the response of containments to internal pressurization conditions beyond design basis that may be useful in supporting development of containment fragility models is provided in Ref. [19].		The experimental programmes were aimed to prove the capability of containment structures under pressure loads beyond conditions defined in the design basis (i.e. beyond design basis conditions) See the term "design basis" is in the IAEA glossary as: **design basis** *The range of conditions and events taken explicitly into account in the design of structures, systems and components and equipment of a facility, according to established criteria, such that the facility can with stand them without exceeding authorized limits.*
159.	ENISS	13	9.17	"End states of the accident progression event tree grouped in a final release category are expected to have similar radiological release characteristics and off-site consequences, […]"	To be consistent with the definition introduced in para 9.16 (see also comment 39).	X			
160.	Finland	2	10.?	Add a statement "Concerning refuelling outage related operating modes, the stage of refuelling (before/after) and the subsequent mixture of newer and older	This is like the paragraph 13.26, which is related to spent fuel pool. Same thing is valid for the reactor core.	X			Added in para 10.17 after sentence "The source term, therefore, could be expressed in terms of the

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				fuel elements should be considered in the definition of the core inventory."					fraction of the initial core inventory of one or more of these groups of radionuclides."
161.	Russian Federation	19	Section 10 ANALYSIS OF UNCERTAINTIES	A plant specific list of uncertain parameters to be varied in the frame of the uncertainty/sensitivity analysis should be derived.At forming the list of parameters for uncertainty analysis, it should not include as parameters a correlation coefficients, model parameters, etc. used in modeling the phenomenology of severe accidents in the corresponding computer codes, established as part of the computer code validation procedure. Otherwise, their variation can lead to completely incorrect results of the uncertainty analysis.	Applicable comments 166.Add new Item.			X	Comment already integrated in para 6.26. See answer to comment 129.
162.	ENISS	14	10(new opening paragraph)	"<u>This Section provides recommendations on release categories specification and source term analysis. 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 in a Level 3 PSA, the characteristics of the environmental source term may need to be more extensive. On the other end of the spectrum, only the frequency of accidents that would result in a large early release may need to be characterized. The following recommendations can therefore be adapted according to the objectives of the PSA</u>."	Proposed a new (opening) paragraph before current para. 10.1. in order to indicate that the extent to which source term analysis needs to be carried out depends on the objectives of the PSA. In some case, source term calculations are not very necessary and risk insights can be obtained just based on the frequency analysis.(note: elements proposed are mainly from previous SSG-4 guide).	X			

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163.	Russian Federation	15	10.2	All potential plant specific release paths should be identified in the accident progression event tree and considered in the corresponding end states. For practical reasons, in accordance with Fig. 1, the end states of the accident progression event tree are generally grouped into release categories (with similar properties regarding releases). The source term analysis is then carried out only for a representative severe accident scenario of each release category. Preliminary list of representative severe accident scenario should be based on severe accident scenario established for identified PDSs (see Item 6.8). The choice of representative scenarios for final list should be justified. It is good practice to carry out sensitivity studies for the choice of representatives scenarios.	Clarification	X			
164.	Sweden	37	10.3 (a)	Hence, the source term analysis in Level 2 PSA involves: Defining the release categories; Grouping of the end states of the accident progression event tree into the defined release categories; Carrying out the source term analysis for the release categories.	Consider having the grouping as a separate bullet.	X			
165.	Egypt	8	Table 7	design extension conditions leakage	The term "beyond design basis conditions" is no longer used in IAEA publications.		X Design basis ~~accident~~ conditions leakage Beyond design basis ~~accident~~ conditions leakage		See the term "design basis" is in the IAEA glossary as: **design basis** *The range of conditions and events taken explicitly into account in the design of structures, systems andcomponents and equipment of a*

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									facility, according to established criteria, such that the facility can withstandthem without exceeding authorized limits.
166.	Russian Federation	16	Table 7, first row	Time frame of the severe accident at which the containment failure/damage	It is proposed to replace the words "the release begins" with "containment failure/damage" in order to be consistent with the wording of the nodal questions from Table 6. In addition, since, as a rule, there is a design leak for a containment, releases into the environment always begin when radioactive medium appears in the containment environment.		X Time frame of the severe accident in which the containment damage/bypass first occurs.		To take account of potential containment bypasses.
167.	Russian Federation	17	Table 7	LOCA outside containment Steam generator tube/tubes or header rupture	It is proposed in the list of values in Table 7 for the attribute "Modes or mechanisms of containment leakage (associated with a time frame)":1) Add: "LOCA outside containment" to account for possible leaks outside the container.2) Change the value "Steam generator tube rupture" to "Steam generator tube/tubes or header rupture" to reflect the accounting of IEs other than the rupture of one SG tube.	X		X	Loss of coolant accident in interfacing system covers LOCA outside containment.
168.	Sweden	38	Table 7	Consider adding Source term: Amount and composition of different radioactive nuclides / nuclide groups Duration: e.g. release during X h	These aspects are missing. Some type of estimate of the duration of time should be included, but could also be explained in qualitative terms.	X			
169.	Sweden	39	10 General	Different release categories may have the same source term (amount and composition).	Consider to have this information somewhere		X Thus, there are many ways of specifying the attributes of a radiological source term, including that different release categories may		Added as part of para 10.12

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							have the same source term (i.e. same amount and composition).		
170.	France	3	10.7	**Some accident scenarios can include several containment failure modes. The analyst should pay attention to quantification of frequency of each containment failure individually in order to comment its importance on the global results**	Some late containment failure mode can be hidden by the first ones in the accident chronology. 10.7 is maybe not the good place in the guide.		X 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		Minor modification for better reading
171.	Japan	10	10.7	In Level 2 PSA, the source term specifies, for a given accident scenario, the quantity of radioactive material released from the plant to the environment and the kinetics of release. Many plant design features and accident phenomena have been recognized to affect the magnitude and characteristics of source terms for severe accidents. <u>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. These plant design characteristics will be the same for all the end states of the accident progression event tree.</u> 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.	Parameters with different properties are included in the sentence after "such as". The classification should be organized and described.			X	There are just examples.
172.	Sweden	40	10.7	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.	Consider changing, including to delete "and the kinetics of release".		X accepted to change "the quantity" to "the amount and composition" as 10.9. In Level 2 PSA, the source term specifies, for a given accident scenario, the quantity amount and		The kinetics was expanded to consider the time and duration of the release, the location (potential energy) and the

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							composition of radioactive material released from the plant to the environment and the timing, location and energy kinetics of the release.		kinetic energy (related to the movement of the release).
173.	Sweden	41	10	General, missing some wording about the need to consider decay.	Suggest to include guidance.		X… Many plant design features and accident phenomena have been recognized to affect the magnitude and characteristics of source terms for severe accidents. 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.		Added as part of para 10.9
174.	Russian Federation	18	Footnote 17 to Item 10.10	The way the attributes are specified is also influenced by the objectives of the Level 2 PSA, for example, whether or not a Level 3 PSA or part of Level 3 PSA will be performed.	Clarification.In footnote 17, it is proposed to replace: "a Level 3 PSA will be performed" with the text: "a Level 3 PSA or part of Level 3 PSA will be performed". It is not always necessary to perform Level 3 PSA in full.	X			
175.	Japan	11	10.15	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 ~~group~~.	The meaning of "group" seems unclear.	X			
176.	Russian Federation	20	10.16	No	The purpose and content of clause 10.16 are not clear. It is proposed either to expand the substantive part of paragraph			X	This is to acknowledge that that radioactive releases have been

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					10.16, including explanations about its purpose, or to delete it entirely.				calculated using dynamic PSA proposing a more realistic results.
177.	Japan	12	10.17 Table	EXAMPLES OF ~~TYPICAL~~ GROUP CATEGORIES FOR ELEMENTS IN RADIOACTIVE MATERIAL	Halogens (oxidized) are not known to be a typical group. They are not modeled in some popular accident progression analysis codes.The word "typical" should be removed, as it seems to be a good thing to mention halogens (oxidized).	X			
178.	Sweden	42	10.26 and 10.27	Consider if these para are possible to delete to avoid repetition.	The general guidance on verification and validation (10.26) and training (10.27) is considered enough and these two para can be deleted.			X	It is important to recall the recommendations with regard to the source term calculations.
179.	Sweden	43	10.28	TABLE 10 at the end of section 10.	Consider adding more specific reference as a help for the reader.	X			
180.	Sweden	44	10.32	"In addition …"	Consider deleting the two first sentences since it is no needed repetition.			X	This is just to recall the requirement.
181.	Sweden	45	10.32	Problem with automatic referencing		X			
182.	Egypt	9	10.34	Uncertainties associated with containment response to design extension conditions lead to uncertainty in respect of the driving forces for radioactive material transport along the pathway to the environment.	The term "beyond design basis conditions" is no longer used in IAEA publications.		X Uncertainties associated with containment response to beyond design basis ~~accident~~ conditions lead to uncertainty in respect of the driving forces for radioactive material transport along the pathway to the environment.		See the term "design basis" in the IAEA glossary as: **design basis** *<u>The range of conditions</u> and events taken explicitly into account in the design of structures, systems andcomponents and equipment of a facility, according to established criteria, such that the facility can withstandthem without exceeding authorized limits.*

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183.	Sweden	46	10.34	Consider if these para are possible to delete to avoid repetition.	Consider deleting, repetition.			X	It is to introduce table 9.
184.	Russian Federation	21	Table 9	No	It is proposed:1) To exclude from Table 9 the phenomenon: "Interaction between hydrogen burn or radicals in flame fronts and airborne radioactive material" or provide more detailed information on it, revealing its essence, as well as indicate computer codes investigating it.3) To provide additional explanations to the phenomenon "Effects of fuel exposure (burnup) on the release rate of radioactive material from fuel". What is meant here by "release rate" – the rate of radionuclides migration in the fuel matrix and their release from the fuel matrix into the gas gap of fuel rods?		X Text modified as: • Effects of fuel exposure (burnup) on the release fraction ~~rate~~ of radioactive material from fuel matrix; • Interaction between hydrogen burn or radicals in flame fronts and airborne radioactive material (e.g. possible resuspension of radioactive deposits);		Modifications proposed for clarification. Table 9 title modified as "Examples of issues…"
185.	Ukraine	11	para.10.9, line 4	Text in brackets: comma should be added between references [48] and [49]	Editorial	X			
186.	Russian Federation	22	10.33	Past and ongoing research programmes have made significant progress towards reducing uncertainty in severe accident source terms (e.g. Refs [53], [54]). Uncertainties associated with the physical processes involved in core damage and core relocation lead to uncertainty in respect of the release of radioactive material from fuel (see Section 6). Uncertainties associated with containment response to beyond design basis accident conditions lead to uncertainty in respect of the driving forces for radioactive material transport along the pathway to the environment (see Section 7).	In order to systematize information on uncertainties analysis and to exclude its repetition in various places of the guide.It is proposed to delete the text "Examples of uncertainties associated with these areas are given in Section 7." from paragraph 10.33, and instead indicate (see Section 7) at the end of the last sentence. In addition, the examples mentioned in the last sentence of paragraph 10.33 are proposed to be given in the appendix table/tables, where it is proposed to move tables 4 and 9 (see comment # 17). At the same time, this information should be supplemented with an approximate list			X	The sentence only indicates the examples presented in section 7.Tables 4 and 9 provide information of possible sources of uncertainties but further details were not considered in the phase of drafting (see answer to comment 134).

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					of specific parameters for which an uncertainty analysis will be performed.				
187.	Russian Federation	23	10.34	The Level 2 PSA should represent the up-to-date knowledge on severe accidents and on fission products behaviour. The assessment of uncertainties can be addressed by carrying out sensitivity studies for the major sources of uncertainty that influence the results of the Level 2 PSA (see also Items 11.25, 11.26). Uncertainties modelling can be also introduced directly in the accident progression event tree (distribution of probability) for their propagation inside the model, while it is possible depending on the PSA tool.	Clarification.It is proposed in paragraph 10.34 at the end of the sentence "The assessment of uncertainties can be addressed by carrying out sensitivity studies for the major sources of uncertainty that influence the results of the Level 2 PSA" to indicate in parentheses: "(see Items 11.25, 11.26)".	X			
188.	Russian Federation	24	Table 10	Fraction of initial core inventory to environment	Clarification It is proposed to clarify the heading from Table 10 "Fraction of core inventory to environment" and use it in the form of "Fraction of initial core inventory to environment".			X	The purpose is to consider the core inventory at the moment of the severe accident.
189.	Russian Federation	25	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 ".			X	The value of the leakage refers to the normal operating conditions (measured by tests), which might be different (higher) than design.
190.	Russian Federation	26	Section 11	For the purpose of general verification of the correctness of the severe accident progression sequences modeling the validation of release categories frequencies sum against the core damage frequency determined from Level 1 PSA (typically core and/or fuel damage frequency) should be done. Justification	AddBy analogy with paragraph 5.3, in order to verify the correctness of the results obtained, it is proposed to include the following in section 11: "For the purpose of general verification of the correctness of the severe accident progression sequences modeling the validation of release categories frequencies sum against the core	X			Added as new para 11.6. Note:To be verified if it is not already covered by text in para 11.7 (former 11.6)

N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
				for any numerical deviations should be given	damage frequency determined from Level 1 PSA (typically core and/or fuel damage frequency) should be done. Justification for any numerical deviations should be given».				
191.	Russian Federation	27	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, 8.17-8.21, 10.32-10.34, 11.17-11.26 include in a separate Section "IMPORTANCE, UNCERTAINTY AND SENSITIVITY ANALYSES". From Sections 5 to 11 provide links to new Section. Provide in new Section subheadings corresponding to Sections 5-11.			X	Paras mentioned in section 5 to 11 before Section "IMPORTANCE, UNCERTAINTY AND SENSITIVITY ANALYSES" where provided to identify sources of uncertainty. The text in para 11.118 to 11.27 provides recommendations on how to treat them. There is no need to create a new section.
192.	ENISS	15	11.1	"The quantification process consists of calculating the frequencies of the end states (i.e. initial release categories) of the accident progression event tree."	To be consistent with the definition introduced in para 9.16. (see also comment 39).			X	The term "initial release categories" and "final release categories" were deleted to avoid confusion, since during the quantification, only release categories are calculated.
193.	Sweden	47	11.4	Consider if these para are possible to delete to avoid repetition.	The general guidance on verification and validation (10.26) and training (10.27) is considered enough and these two can be deleted.		X 11.4. The probabilistic quantification of the Level 2 PSA should be carried out using a suitable computer code that has been fully validated and verified.		It refers to codes for probabilistic calculations
194.	Sweden	48	Table 11	TABLE 11. MITIGATION PERFORMANCE MATRIX	Propose another word which is more broader	X			

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195.	Japan	13	11.21(a)	Incompleteness uncertainty. The overall aim of a Level 2 PSA is to assess the possible scenarios (sequences of events) that can lead to releases of radionuclides, mainly those scenarios modelled in the Level 1 PSA. However, there is no guarantee that this process can ever be complete and that all possible scenarios have been identified and properly assessed. This potential lack of completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify. It is not possible to address this type of uncertainty explicitly. However, extensive peer review can reduce this type of uncertainty, for example by verifying the adequacy of the sequence consisted by cutsets, correctness of the input parameters, and assumption of human error, so the extensive peer review of 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.	To clarify the reason that the extensive peer review can reduce uncertainty.		X…This potential lack of completeness introduces an uncertainty in the results and conclusions of the analysis that is difficult to assess or quantify. It is not possible to address this type of uncertainty explicitly. However, extensive peer review can reduce this type of uncertainty, for example by verifying the adequacy of the sequence consisted by cutsets, correctness of the input parameters, and assumption of human error, so the Level 2 PSA should have extensive peer review….		Simplification of the text proposed for better reading.
196.	Sweden	49	11.21	Problem with automatic referencing		X			
197.	Russian Federation	28	11.25	Parameter/event/phenomenon specific sensitivity analysis may be used instead of comprehensive uncertainty analysis. Sensitivity analysis is a useful tool to guide the selection of dominant sources of uncertainty. In this case, it is allowed	Clarification and compliance assurance.It is proposed,1) To supplement paragraph 11.25 with the following: "In this case, it is allowed to select parameters / events /phenomena subject to sensitivity analysis in		X 11.26. Parameter/event/phenomenon specific sensitivity analysis may be used to supplement a more comprehensive uncertainty analysis. Sensitivity analysis is a useful tool	X	Second proposed sentence "In this case…" rejected since it makes text less readable. Sentence "Example …" is not deleted since the

N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
				to select parameters/events/phenomena subject to sensitivity analysis in accordance with the established selection criteria	accordance with the established selection criteria".2) Paragraph 11.26 indicates the possibility of performing sensitivity analysis instead of uncertainty analysis. To ensure compliance with the information from paragraph 11.25 of paragraph 11.26, as well as to take into account all aspects subject to uncertainty/sensitivity analysis, the following wording of the first sentence of paragraph 11.25 is proposed: "Parameter/event/phenomenon specific sensitivity analysis may be used instead of comprehensive uncertainty analysis".3) Delete the sentence "Example areas of uncertainty related to the progression of severe accidents are listed in Table 4", because paragraph 11.25 refers to sensitivity analysis.		to guide the selection of dominant sources of uncertainty. Example areas of uncertainty related to the progression of severe accidents are listed in TABLE 4.		identification of the sources of uncertainty is used for the sensitivity analysis.
198.	Sweden	50	12.2	"…interim reports, the PSA reference report including a rather comprehensive summary (usually split up in several files), and the SAR chapter or similar."	The use of the term "external report" is not clear. SAR Chapter is usually what is delivered or made available to the regulator. There may also be other documentation for the public in some countries?		X… and the reference ~~final external~~ report of the PSA, which might be or not in addition to the Safety Analysis Report.		Text modified for clarity, since the PSA report might not be fully in the SAR.
199.	Sweden	51	12.3	The PSA reference report (s) should include all the information needed to reconstruct the results of the study. The results of ..	Prefer not to use the term "external report".	X			
200.	Sweden	52	12.8	Consider replacing "contributory" with "supporting"	The term contributory is used in a consistent manner.	X			
201.	Japan	14	12.11	<u>The results of the PSA may be compared with probabilistic safety criteria for Level 2 PSA, if these have been set. Available probabilistic safety criteria and/or goals vary considerably among Member States,</u>	Missing place.			X	Annex number modified.

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				but the most common risk metrics for Level 2 PSA include criteria and/or goals for the frequency of a large early release and the maximum tolerable frequency of releases of various magnitudes (see paras 2.15 to 2.18 and Annex IV). While the threshold for large early release frequency represents a point estimate frequency for a particular unacceptable release, the maximum tolerable frequency of releases of various magnitudes expands this concept across the full range of possible releases.					
202.	Ukraine	5	12.21	Incorrect reference to Annex III should be replaced with reference to Annex II	Editorial	X			
203.	Sweden	53	12.21	Check appendices / Annexes and provide accurate numbering and referencing	Both appendix and annex are used, why the referencing is sometimes difficult to follow. Consider clarification on the use of these concepts to this guide.	X			
204.	Sweden	54	13.1	Delete " .. potential limited mitigation capabilities"	Potential limited mitigation capabilities is a finding in a level 2 PSA that might be easy to identify even with a brief study. However, not really an argument not to do the work.			X	Potential limited mitigation capabilities when a severe fuel damage occurs in the SFP is not a finding of Level 2 PSA, but a input condition which leads to that there is no need to perform Level 2 PSA.
205.	Sweden	55	13.2	Replace "water bodies" with "water sources"	The term water bodies is not a widely used term, please consider to use *water sources*, viewed as a better choice.	X			

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206.	Japan	15	13.2 & 13.17	13.2. This section focuses its recommendations for the development of Level 2 PSA when the spent fuel pool is located inside a building capable to ensure the confinement function in severe accident conditions. If not, one practice has been to consider in Level 2 PSA that accidents involving damage of fuel stored in the spent fuel pool lead directly to large radioactive releases. A complement to this practice is to proceed with an analysis aiming at substantiating the capabilities for crediting some fission product retention in buildings or water bodies in severe accident conditions. Spent fuel pool criticality is not considered because it is not likely due to the amount of fissile material in the SFP, as well as its geometrical configuration and presence of neutron absorbing material.~~13.17. In general, spent fuel pool criticality is not likely due to the amount of fissile material in the SFP, as well as its geometrical configuration and presence of neutron absorbing material.~~	Para. 13.17 describes a general remark, thus it should be included in para. 13.2.			X	Recommendations on what not to do are also valid and the topics covered in 13.17 are related to specific accident progression which might not be part of the analysis, however they need to be justified. (See resolution of comment 214)

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207.	Sweden	56	13.4	Strange wording " For example, the location of the pool determines whether an accident progression event tree is necessary to be developed or whether other factors that could reduce the source term could be taken into consideration."	Consider deleting the example. What is meant by "other factors"? Other factors should also go into an APET?		X For example, location of the pool determines whether an accident progression event tree is necessary to be developed or whether other factors that could reduce the source term could be taken into consideration (e.g. possibility to close the containment, (i.e. if the spent fuel pool is located inside the containment), availability of the ventilation system and of the spent fuel cooling system).		Examples were proposed to explain other factors.
208.	Egypt	10	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 SSG-3 (Rev.1) (para 10.2-10.6)	Para. 10.2 – 10.6 in SSG-3 are for use and applications of PSA			X	The paras are in the new SSG-3 as approved by the CSS.
209.	Japan	16	13.7	If the spent fuel pool PSA and the reactor PSA are combined, the PDS should consider combined reactor and spent fuel pool PDS. Reactor accident sequences can impact the spent fuel pool, for example containment venting could accelerate boiling of the water in the SFP in case SFP locates in the containment. In addition, reactor accident sequences that do not result in Level 1 reactor core damage events may impact the mitigation actions for the spent fuel pool accidents and may have to be considered for inclusion in the PDS.	It should be specified that this description is for the special design.		X could accelerate boiling of the water if the SFP is located inside the containment….		For better reading.

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210.	ENISS	16	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."	Text improvement to recall the scope of this recommendation	X			
211.	Russian Federation	29	13.8	To support Level 2 PSA, 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 behaviour (fuel burnup, decay heat, cladding behaviour, etc.), fuel assembly and rack degradation (zirconium clad reaction and hydrogen generation, zirconium fire, corium-concrete interaction, if considered), 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.	Clarification. Taking into account all factors.In order to take into account all factors affecting the severe accident progression in SFP, it is proposed to replace in paragraph 13.8 the phrase "on the fuel assemblies arrangement and burn-up" with "on the fuel assemblies arrangement, burn-up and storage time".	X			
212.	Sweden	58	13.8	Severe accident phenomena to consider in this analysis includes heat transfer within the pool, fuel racks, and to surrounding walls, fuel behaviour (fuel burnup, decay heat, cladding behaviour, etc.), fuel	Editorial	X			

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				assembly and rack degradation (zirconium clad reaction and hydrogen generation, zirconium fire, and corium–concrete interaction, ~~if considered~~), 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 and burn-up in the spent fuel pool.					
213.	ENISS	17	13.12	"Depending on the plant configuration (spent fuel pool in or outside the reactor containment building), severe accident analysis should consider the interactions between the reactor and the spent fuel pool: a reactor accident can have impact on or induce a spent fuel pool accident and vice versa. From this analysis, some additional accident scenarios (involving both reactor and spent fuel pool) could be built in the Level 2 PSA if not already considered in the Level 1 PSA, such as the following: […]"	General text improvement	X			
214.	Russian Federation	30	13.17	In general, spent fuel pool criticality is not likely due to the amount of fissile material in the SFP, as well as its geometrical configuration and presence of neutron absorbing material. Nevertheless, the issues of criticality in SFP should be addressed in the Level 2 PSA documentation.	Clarification. It is proposed to add in paragraph 13.17 the text: "Nevertheless, the issues of criticism in SFP should be addressed in the Level 2 PSA documentation."	X			
215.	ENISS	18	13.19	"If not screened out, dedicated analysis should be performed to address in the Level 2 PSA the consequences of accidents during fuel transfer operations between the spent fuel pool and the reactor. Typical accidents to be	Accidents during fuel transfer operations are to be considered in the PSA if they have not been screened out. Text improvement to be more general.	X			

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				considered are related to fuel uncovering due to the loss of spent fuel pool cooling system caused, for example, by a station blackout or effects due to external hazards (e.g. a seismic event)."					

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216.	USA	10	13.19	Dedicated analysis ~~**should be performed**~~ to address fuel transfer operations between the spent fuel pool and the reactor **should be considered.**	Depending on the design, such analysis for fuel transfer may not be necessary, as it may be bounded by other scenarios.	X			
217.	Egypt	11	14	**LEVEL 2 PSA FOR MULTI-UNIT NUCLEAR POWER PLANTS**	In IAEA publication, the most common term is muli-unit rather than multiple unit.			X	The technical editors made this change. The final terminology will be updated with the one accepted for the SSG-3 (Rev. 1) (DS523).
218.	Sweden	58	14.2	…are not fully addressed.	Editorial	X			
219.	Sweden	59	14.3(b)	Correlated or shared ~~of~~ SSCs…	Editorial	X			
220.	Japan	17	14.3(c)	Impact of consequences induced by a unit with a severe accident on the other units (e.g. additional fuel melt accidents).	The word of "additional fuel melt accidents" is unclear.		X(c) Impact of consequences induced by a unit with a severe accident on the other units (e.g. ~~additional~~ fuel melt accidents happening in another unit).		New text proposed to clarify what "additional" means.
221.	ENISS	19	14.5	"The selection of topics of interest should be such that their treatment will not induce excessive complexity in the development of the Level 2 PSA for multiple unit nuclear power plants. Therefore, according to the selected topics of interest it should be possible to not develop all the elements of the Level 2 PSA as recommended in paras. 14-6 to 14-31, but to have a more straightforward and simplified approach to capture the impact of multiple units Nuclear Power Plants on the PSA insights. In some cases, an approach based on the post-processing of the single unit L2 PSA results could be sufficient to obtain relevant insights."	Proposal to add a new text to offer the possibility to not develop a multi-unit Level 2 PSA model but to have a simplified approach to assess the associated risk (see example of 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	Paras 14.4, 14.22 and 14.25 already consider the possibility to develop a simplified Level 2 PSA.

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222.	Egypt	12	14.6	Recommendations provided in paras 4.1 - 4.18 related to plant familiarization	Paragraphs considering plant familiarization start from: 4.1 – 4.18.	X			
223.	Sweden	60	14.6	Strange referencing paras 4-4.18	Editorial	X			
224.	ENISS	20	14.8	"Traditional risk metrics used in PSA for a single unit site (e.g. large release frequency) could be used as far as possible in order to express the risk profile in the context of multiple unit nuclear power plants for corresponding decision-making (see paras 2.16-2.18). When relevant, these traditional risk metrics could be adapted in specific multi-unit risk metrics such as conditional probability of large releases from several reactors knowing large releases from one reactor of a unit on a multi-unit site."	In general, specific metrics are introduced for MUPSA (as an adaptation of the usual risk metrics used for a single unit).	X			
225.	Sweden	61	14.14	…in Sections 6 and ~~,~~ 7 ~~and 8~~.	Severe accident phenomena not discussed in section 8?			X	It refers to human actions in a severe accident for the multi unit context

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226.	Sweden	62	Subheading before 14.16	…NUCLEA~~E~~R POWER PLANTS	Editorial.	X			
227.	ENISS	21	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."	Related to the previous comment for para. 14.5, this problem mainly concerns the development of a full multi-unit Level 2 PSA model.	X			
228.	Russian Federation	31	15.1	(h) Development of a list of severe accident scenarios to be addressed in the NPP design.	As a separate bullet of application of the Level 2 PSA, it is proposed: "Development of a list of severe accident scenarios to be addressed in the NPP design".	X			
229.	ENISS	22	15.3	"The scope of the Level 2 PSA, as stated in para. 2.7, should be commensurate with its intended uses and applications, and based on the equivalent scope of Level 1 PSA. A full scope of Level 2 PSA is most suitable for a large number of uses and applications, with due considerations given to the uncertainties on key parameters and limited strength of knowledge on some data and assumptions that could impact the PSA results and insights. Since the Level 2 PSA relies on the Level 1 PSA model, this ~~requires~~	As stated in the Chapter in SSG-3 on Level 1 PSA application, it is possible to consider a limited scope of the Level 2 PSA for some applications. It could be acceptable to reduce the scope of the PSA when the uncertainties on key parameters are too important or the strength of knowledge on some data and assumptions too limited to characterize PSA sequences and derive PSA insights.			X	The proposed text is adding misleading messages and the concept of the scope commensurate with applications is already in the original text.

N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
				should require that the Level 1 PSA: (a) Includes an as comprehensive as possible 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 refueling (if not screened out).In any case, when the risk insights are to be derived from a Level 2 PSA that has a smaller scope than the full scope described in this paragraph (e.g. not all initiating events and hazards considered), this should be recognized in applying the insights from the PSA."					
230.	Ukraine	12	para.15.12, footnote 22	The second sentence in footnote 22 basically repeats information in para.15.13 and can be deleted	Editorial	X			
231.	USA	11	15.17	Consideration should be given to making improvements to the features provided for the prevention or mitigation of severe accidents in order to reduce ~~**those**~~ contributions to the overall risk **of sequences** ~~that have~~ **with** the highest risk significance	Text was confusing, suggest wording changes for improved text clarity.	X			
232.	Japan	18	15.32	For a Level 2 PSA that is to be used for emergency preparedness and response, the releases considered should be accurately specified in terms of isotopic composition ~~and~~, amount and timings of radioactive material released (i.e. source terms), as well as in terms of relevant additional attributes (see TABLE 7 in Section 10).	Time information is considered very important in the field of emergency preparedness.		X …amount and timing of…		For better reading.
233.	Germany	12	Annex I	RFERECES TO ANNEX I~~I~~	Editorial	X			

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234.	Japan	19	ANNEX ITABLE I-1I-9.TABLE I-2TABLE I-3I-10.TABLE I-4	It is preferable to delete everything from the last sentence in I-8. "Major codes of this type are summarized in Table I-1." to I-10 of ANNEX I, including Table I-1, I-2, I-3, and I-4. Also related reference of ANNEX I, from reference 22 to 36, should be deleted.	The update of analysis codes for level 2 PSA is too frequent to prevent the obsoleteness of the information. Listing the names of analysis codes could make the readers refer to outdated information. Thus, it is not desirable to describe specific analysis codes.			X	This Safety Guide aims at providing the information of the codes that is currently valid. This information is important for newcomers. Additionally, if updated version of the codes will exist, they might be introduced in future versions/revisions of this Safety Guide.

235.	Japan	20	ANNEX III1.22.2.16. 12.21.	ANNEX III should be deleted.The sentences related to ANNEX III in paras. 1.22 and 2.16, and all sentence of 12.21 should be also deleted.1.22. Sections 2–12 of this Safety Guide provide recommendations on the performance of Level 2 PSA, with each section corresponding to a major procedural step in Level 2 PSA as shown in Fig. 2. Section 13 provides recommendations on the performance of Level 2 PSA for the spent fuel pool. Section 14 provides recommendations on the performance of Level 2 PSA for a site with multiple nuclear power plants (also known as a multi-unit site). Section 15 provides recommendations on the uses and applications of a Level 2 PSA. The Appendix gives an overview of human reliability analysis in Level 2 PSA. Annex I discusses various types of computer code available for simulation of severe accidents and PSA studies. Annex II presents a sample outline of documentation for a Level 2 PSA. ~~Annex III provides information on the common risk metrics used in Level 2 PSA with examples from several Member States.~~2.16. Large release frequency and large early release frequency are the most common risk metrics used in Level 2 PSA, but there is variation among Member States ~~(see Annex III)~~. ~~12.21. A sample outline for the documentation for a Level 2 PSA is provided in Annex III.~~ In addition, all reference documents in ANNEX III.	Describing safety goals only in the guide of Level 2 PSA (DS-528, revision of SSG-4) is not suitable since the guide of Level 1 PSA (DS523, revision of SSG-3) does not provide an appendix or an annex about the safety goals. At the same time, we consider that the content of ANNEX III is beneficial and productive for Member States. Accordingly, it is desirable to provide these information as a TECDOC or a Safety Report.			X	The understanding of risk metrics and probabilistic goals for Level 2 PSA has been raised several international meetings and it was included in the DPP approved by the NUSSC. The Annex III came as a proposal to avoid having numbers in the body of the text.Annex III provides examples with reference to risk metrics and probabilistic goals for Level 2 PSA in some Member States. Further details are provided in a TECDOC currently under preparation.The definition of probabilistic safety goals for Level 1 PSA has more consensus and understanding, therefore the revision of the Safety Guide on Level 1 PSA did not foresee the need for including an annex about the examples.

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236.	Iran, Islamic Republic of	8	ANNEX III Russian Federation	The release of radioactive substances into the environment during an accident at NPP, where it is necessary to carry out measures to protect the population on the border of the planning zone for protective measures in the initial period of the accident.	The term "initial period of the accident" to be more clarified from the points of accident phases and progress."The planning zone for protective measures" to be more clarified.			X	It is important to provide further details related to these abstract terms however, the quoted text is as presented in the reference from Russian Federation.
237.	Russian Federation	32	Annex III, Table III-1, Russian Federation	The release of radioactive substances into the environment during an accident at NPP, when in case of exceeding established criteria for radiation doses it is necessary to implement measures to protect the population within the initial stage of the accident (up to 10 days) on the border of the protective actions planning zone and outside it. It should be noted that established frequency of release it is not a safety goal, it is safety target.	Clarification of information for Russian Federation.Proposed:1) Add information that protection measures are implemented only if the criteria for radiation doses established in the radiation safety standards are exceeded.2) Specify the initial period of the accident – 10 days.3) Indicate that this is not a safety goal, but a safety target.	X			
238.	Ukraine	1	Annex III Table III-1	Ukrainian regulation NP 306.2.141-2008 "General Safety Provisions for Nuclear Power Plants" defines the following safety criteria and goals for LRF:criterion / goal for existing plants: < $1{\cdot}10^{-5}$ / $1{\cdot}10^{-6}$ 1/r.y.;criterion / goal for new plants: < $1{\cdot}10^{-6}$ / $1{\cdot}10^{-7}$ 1/r.y.This information can be added to the last column with corresponding reference	Missing information		X criterion / goal for existing plants: < $1{\cdot}10^{-6}$ 1/r.y.;criterion / goal for new plants: < $1{\cdot}10^{-7}$ 1/r.y.		Only the values related to large release frequency are presented.

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239.	ENISS	23	Annex III Table III-1	*Large release frequency risk metrics DefinitionFRANCE*"For new nuclear power plants: protective measures for the public should be very limited in terms of extension and duration, meaning no permanent relocation, ~~neither~~ **no** evacuation ~~nor sheltering~~ needed outside of the immediate vicinity of the plant site, neither sheltering nor ~~and no~~ long-term restriction of food consumption outside the vicinity of the plant site. Consequently, these accidents should not lead to neither contamination of large areas nor long-term environmental pollution."	The ref. [III-17] (Guide de l'ASN n°22) applies to new nuclear. No sheltering is only required outside of the vicinity of the plant site (and not outside of the **immediate** vicinity of the plant site).		X Primarily for new nuclear power plant designs: Protective…		The text in section 1.3 (of the reference [III-17]) relative to the scope of application mentions: "With a primary focus on the design of new design of new PWRs, the recommendations of this guide can also be used as a also be used as a reference, for the research of improvements to be made to existing reactors,…
240.	Russian Federation	33	Annex III, Table III-2, Russian Federation	- Term LERF is not defined.	Add information for Russian Federation.It is proposed to indicate that the term LERF is not defined in the Russian Federation.	X			
241.	ENISS	24	Editorial	- page number return to 1 after table of contents and in the middle of para. 2.2"**Error! Reference source not found"** found in paras. 3.23 ; 5.5 ; 5.6 ; 5.10 ; 6.1 ; 6.14 ; 7.3 ; 7.10 ; 10.32 ; 11.21reference to a non-existent "Section 0" found in paras. 6.1 ; 11.15reference to a non-existent "para. 0" found in para. 7.25		X			
242.	ENISS	25	Editorial 2.30	"Later changes can be addressed in the framework of the periodic safety reviews, as part of a living PSA programme, as described in paras. ~~**2.19**~~2.19–2.22."	Para. 2.19 is repeated twice	X			
243.	ENISS	26	Editorial 3.5	"Paragraphs ~~**3.7**~~3.6-3.7 provide recommendations […]"	Para. 3.7 is repeated twice	X			

Table of resolution of NUSSC Members' comments for Development and Application of Level 2 Probabilistic Safety Assessment for Nuclear Power Plants, STEP 7 (DS528) NUSSC 55th Meeting

N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
244.	ENISS	27	Editorial 4.12	(…) from the containment. IAEA Safety Standards Series (…)	Dot	X			
245.	ENISS	28	Editorial 4.13	"Paragraphs ~~0~~**4.14**-4.15 provide recommendations on […]"	Error in the number of paragraph called.	X			
246.	ENISS	39	Editorial 6.1	(…) needing to be included (…)	-	X			
247.	ENISS	30	Editorial 6.22	Last item on list should be (e)	List restart in p. 31	X			
248.	ENISS	31	Editorial 6.27	(…) calculated key variables (…)	Adjective before noun	X			
249.	ENISS	32	Editorial 7.25	~~(see para 0)~~	Provide right ref	X			

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N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
250.	ENISS	33	Editorial 10.1 note 14	~~As defined in Ref. [47]~~	Double ref. for same definition. Also, attention to the closing quotes (missing)		X footnote 14 modified as:The term 'source term' is to be understood as defined in the IAEA Nuclear Safety and Security Glossary [46] as "The amount and isotopic composition of radioactive material released (or postulated to be released) from afacility. Used in modelling releases of radionuclides to the environment, in particular in the context of accidents at nuclear installations or releases from radioactive waste in repositories." . In addition, other definition providing more details is "The characteristics of a radionuclide release at a particular location including the physical and chemical properties of released material, release magnitude, heat content (or energy) of the carrier fluid, location relative to local obstacles that would affect transport away from the release point, and the temporal variations in these parameters (e.g., time of release, duration, etc.)", as defined in Ref. [47].		The definition in IAEA Safety Glossary is not too much detailed, that is why the second definition is added with the reference.
251.	ENISS	34	Editorial 10.3 (a)	(…)which might include ~~the~~ grouping the end states (…)	-	X			
252.	ENISS	35	Editorial 11.15	…as discussed in section ~~0~~…	Hyperlink to fix	X			
253.	ENISS	36	Editorial 15.8	…(see Refs [66], [67]).	-	X			

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N	MS	Comment No.	Para/ Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
254.	ENISS	37	Editorial 15.12 footnote 22		Unnecessary – the definition of the qual. Obj. for future plants described at next para.		X footnote 22 modified as:According to Ref. [7], the objective for large off-site releases requiring short term off-site response is 1x10-5 per reactor-year for existing plants.		The second part of the text was deleted since it is already quoted in para 15.14

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<table>
<tr><th>N</th><th>MS</th><th>Comment No.</th><th>Para/Line No.</th><th>Proposed new text</th><th>Reason</th><th>Accepted</th><th>Accepted, butmodified as follows</th><th>Rejected</th><th>Reason formodification/rejection</th></tr>
<tr><td>255</td><td>Republic of Korea</td><td>1</td><td>TABLE III–2</td><td><table>
<tr><th>Member State</th><th>Reference</th><th>LERF risk metrics definition</th><th>Safety goal frequency, 1/r.y.</th></tr>
<tr><td>…</td><td>…</td><td>…</td><td>…</td></tr>
<tr><td>Korea, Republic</td><td>[III-7]</td><td>~~100 TBq of Cs-137~~The frequency of those accidents leading to significant, unmitigated releases from containment in a time frame prior to effective evacuation of the close-in population such that there is a potential for early health effects.</td><td>< $1*10^{-5}$ for operated NPP< $1*10^{-6}$ for new NPP</td></tr>
</table></td><td>100 TBq of Cs-137 is wrong definition. New text is a right definition of LERF in Korea.</td><td>X</td><td></td><td></td><td></td></tr>
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