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MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Finland	1.	General	Please check the use of the term radioactive material.	It has been used 9 times in the document. at least for the Table 1 level 5 objective, para 3.40 and para. 4.4 deal with radioactive releases and in line with the IAEA Glossary term radioactive substance should be used.		X		The correct use of the terms "radioactive material" and "radioactive substances" has been updated according to suggestion of technical editors and in compliance with the IAEA glossary.
Germany	1	1.9	~~This Safety Guide considers the assessment of the independence of structures, systems and components implemented at different defence-in-depth levels in a general manner.~~ This Safety Guide considers the assessment of the degree of independence between levels of defence in depth and, in a general manner, the assessment of independence of structures, systems and components implemented at different defence-in-depth levels.	We cannot retrace the origin of the change in para. 1.9. The Ukrainian comment from Step 11, referred in the Version for the Silence Procedure and accepted before the 53. NUSSC Meeting, was: "This Safety Guide considers the assessment of the independence of defence-in-depth levels and, in a general manner, the assessment of independence of structures, systems and components implemented at different defence-in-depth levels". The Canadian comment from Step 11, accepted before the 53. NUSSC Meeting reads: "This Safety Guide considers the assessment of the degree of independence between levels of defence in depth and, in a general manner, the assessment of independence of structures, systems and components". As far as we can see para 1.9 was not a subject of discussion during/after 53.			X	The comments mentioned were prior to the NUSSC 53rd meeting. After that, the change in para 1.9 was proposed by the technical editor in the version presented for the NUSSC 53rd meeting after collecting all NUSSC Members comments. Event though, this change was presented in the version discussed during the NUSSC 53rd meeting, none commented it during the meeting. The text proposed is too complicated, repetitive and there is not such an assessment of the degree of independence of DiD levels in the DS508. Therefore, it is proposed to keep the text as it is proposed in the version discussed during the NUSSC 53rd meeting and presented for the silence procedure since it is simple and represent the real content of

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				NUSSC Meeting. We would like to ask you kindly to restore the previous, original text.				the DS508.
Germany	2	1.12	Section 2 sets out the requirements in SSR-2/1 (Rev. 1) [1] that govern the approach to design of nuclear power plants relating to prevention of radiological consequences, on which the recommendations in this Safety Guide are based. Section 3 provides recommendations on the implementation and assessment of design extension conditions within the concept of defence in depth, and on independence of ~~safety provisions considered for~~ the levels of defence in depth. Section 4 provides recommendations on the application of the concept of practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release. Section 5 provides recommendations on the implementation of design provisions for enabling the use of non-permanent equipment for power supply and cooling.	As these changes are technical follow-ups of changes for para. 1.9, we would like to ask you kindly to restore the previous, original text here as well.			X	Even though the titles of sections in chapter 3 do not mention explicitly the independence of safety provisions, the recommendations aim to the independence of safety provisions required at different levels of DiD. The proposed text for para 1.9 was rejected.
ENISS	1	2.8/2.9	Harmful radiological consequences to the public can arise only from the occurrence of uncontrolled accidents. Therefore, recommendations in the following sections are devoted to the	For clarification: We do not understand the red marked additional text : 1.It wrongly says (therefore) that radiological consequences arise from DEC only. Uncontrolled DBC are not			X	The text added was in version step 9 addressing comments by MS. For simplification was deleted by technical editor before NUSSC and restored by TO for better understanding of

MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			implementation and assessment of design extension conditions within the concept of defence in depth and the complementary need for demonstration of practical elimination of plant event sequences that could lead to an early radioactive release or a large radioactive release.	covering all DEC and are source of radiological consequences. 2.We understand the need to incorporate a change for Annex I to state that the consideration of practical elimination may vary from MS, but in that case "the complementary need for demonstration of practical elimination" has to be removed as this is not shared by all MS.				the text in para 2.8. 1. It does not say that radiological consequences arise from DEC only. The first sentence mentions "uncontrolled accidents" without specifying DBC or DEC. It does say the recommendations here after are devoted to both the implementation and assessment of DEC and the PE. 2. The recommendations proposed aim at providing elements that help to the demonstration of PE concept with the objective to be accepted by all MSs.
UK	1	3.5	Original wording: "Design extension conditions without significant fuel degradation could be understood as those representative event sequences involving either a single initiating event of very low frequency, or an anticipated operational occurrence or infrequent faults of design basis accident combined with multiple failures, which are considered in the design in order to prevent reactor core melt and melting of fuel stored in the spent fuel pool." Change to read "Design extension conditions without significant fuel	Sentence is too long and complicated. Also, Annex II Table II-1 of SSG-2 states that 'infrequent faults' is an alternative to the term 'design basic accidents' used by some MS, whereas as originally worded it reads as though it's being used as a 'tier' (sub-set) of the higher frequency design basis accidents. We suspect the UK is the MS referred to in Annex II which uses 'infrequent faults' as an alternative to DBA, but in our lexicon the original text would mean the opposite of what is intended.	X	Footnote 7 is modified as: Infrequent faults term is used in Table II-1 in Annex II of SSG-2 (Rev.1) [9], however some Member States may use a different definition.		

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			degradation could arise from event sequences involving a single initiating event of very low frequency. Additionally, design extension conditions could arise from event sequences involving anticipated operational occurrences or higher frequency design basis accidents, combined with multiple failures.". Footnote 7 can remain.					
Germany	3	Footnote 9 (new and old)	~~Such safety features are understood as additional safety features for design extension conditions, or as safety systems with an extended capability to prevent severe accidents (see para. 5.27 of SSR-2/1 (Rev. 1)) [1].~~ Footnote 8: Such safety features are understood as additional safety features for design extension conditions, or as safety systems with an extended capability to ~~mitigate~~ prevent the consequences of severe accidents (see para. 5.27 of SSR-2/1 (Rev. 1)) [1]. Footnote 9: Such safety features are understood as additional safety features for design extension conditions, or as safety systems with an extended capability to mitigate the consequences of severe accidents or to maintain the integrity of the containment (see para. 5.27 of SSR-2/1 (Rev. 1)) [1].	The draft version after 53.NUSSC has two footnotes in Table 1, No. 8 to Level 3b and No. 9 to Level 4 (reading from the left side of the Table). Both were correct and both were referring to Para 5.27 SSR-2/1 (Rev.1), which states that the safety features might be required for: - design extension conditions, or - extension of the capability of safety systems to prevent, - or to mitigate the consequences of, a severe accident, - or to maintain the integrity of the containment. We have not found a request of NUSSC Members to delete footnote 9 and would like to suggest to restore both footnotes, but with sight textual changes.	X	Footnotes are numbered 9 and 10 in the last version.		
Germany	4	3.8	Operational states comprise two sets of plant states: normal operation and	We agree with the introduction of the definition for "anticipated operational		X		Maintenance and testing are not really normal modes of

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			anticipated operational occurrences. Modes of normal operation include startup, power operation, shutting down, shutdown, maintenance, testing and refuelling and are defined in the documentation governing the operation of the plant (e.g. the operational limits and conditions11). Anticipated operational occurrences12 are deviations from normal operation that could be reached by the occurrence of a postulated initiating event involving a failure to prevent an abnormal operation or an equipment failure either expected to happen during the operating lifetime of the plant.	occurrences" and find it useful. However, deleting of "maintenance and testing" from the first part of para is not in-line with IAEA Safety Glossary 2018 and with SSG-61 (Format and Content of the Safety Analysis Report for Nuclear Power Plants), "Modes of normal operation of the plant", para. 3.1.11; we propose to return to the previous wording. Additionally, we guess "either" should be deleted, as it is redundant.	X	Operational states comprise two sets of plant states: normal operation and anticipated operational occurrences. Modes of normal operation include for example startup, power operation, shutting down, shutdown, and refuelling and are defined in the documentation governing the operation of the plant (e.g. the operational limits and conditions11).	X	operation. Therefore, the proposed text strives to avoid technical contradiction while keeping compliance with the IAEA Safety Glossary 2018 edition.
UK	2	3.14	Original wording - "The safety systems should be designed to mitigate postulated initiating events considered for design basis accidents as challenges to the fulfilment of the safety functions or challenges to the barriers." Change to read – "The safety systems should be designed to mitigate postulated initiating events considered for design basis accidents by ensuring that safety functions can be delivered and barriers are maintained."	Doesn't make sense as written, simplification to wording.		X The safety systems should be designed to control postulated initiating events considered for design basis accidents by ensuring that safety functions can be fulfilled, and barriers are maintained.		Proposed text change "control" instead of "mitigate" by France. Proposed text change "fulfilled" instead "delivered" by TO and technical editor.
France	1	3.14	cannot support the additional word "infrequent and limiting fault" which are not clear.	As mentioned in DS508, they come from an annex of SSG-2 thus are not part of SSG-2, thus they shall not be mentioned as integral part of DS508. That could be easily editorially solved	X			

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France	2	3.14	The safety systems should be designed to ~~mitigate~~ control postulated initiating events considered for design basis accidents …	Editorial to be consistent with SSR-2/1 and other parts of the DS 508	X			See also UK comment 2.X
Germany	5	3.14	Accidents conditions are not expected to occur during the lifetime of the plant. The most frequent accident conditions, which might occur, are categorized as design basis accidents and should have an expected frequency typically below 10-2 per reactor-year. …	It is a good idea to avoid "postulated initiating events" and possible confusions, related to this issue. Our suggestion is an editorial one, to make a transfer between the statements "accident conditions are not expected to occur" and just after that following "the most frequent accident conditions" smoother.	X			
Germany	6	3.17	To meet the requirements presented in paras 3.15 and 3.16, two separate categories of design extension conditions ~~may~~ should be identified[15]: design extension conditions without significant fuel degradation[16] and design extension conditions with core melting.[17] For colleagues, who prefer "may", we suggest to resolve this issue with a separate sentence in an additional footnote – such a solution has been applied in various IAEA Safety Guides before.	We insist on "should". Please keep in mind that "should"-formulations, related to design extension conditions without significant fuel degradation and to design extension conditions with core melting, are fixed already in a number of IAEA Safety Guides. SSG-2, Rev.1 is one of them. Hence "should" in para 3.17 is also a question of consistency, in addition to the question of safety for the major types of NPPs worldwide. We also understand colleagues, which are holding technologies, where core melting is rather unlikely owing to the physical-construction reasons, and suggest to resolve the issue with a separate sentence in an additional footnote – such a solution has been			X	The text proposed in this safety guide aims at compliance with requirement 20 of the SSR-2/1 (Rev.1) where there is no clear distinction between two design extension conditions as proposed in approach 2 of table 1 in this draft safety guide. This wording was particularly discussed during the 53rd NUSSC meeting therefore, the final text needs approval of all NUSSC Members.

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				applied in various IAEA Safety Guides before.				
France	3	3.19	~~A difference between design basis accidents and design extension conditions without significant fuel degradation is established primarily based on their frequencies of occurrence (see~~According to Requirement 13 of SSR 2/1 (Rev.1) [1]~~)~~., plant states shall be identified and shall be grouped into a limited number of categories primarily on the basis of their frequency of occurrence at the nuclear power plant.	requirement 13 of SSR-2/1 is mentioned but is not exactly quoted. As a consequence, the sentence is not fully consistent with French practice (this is not "the difference"). That could be solved by quotation of SSR-2/1.	X			
Finland	2.	3.21 (c)	"… overall limits and criteria…"	Remove "acceptable" as it is not needed.		X …overall limits or criteria related to…		
France	4	3.21 (c)	…overall acceptable limits or criteria related to the radiological…	5.31 is for practical elimination, thus, not for this chapter 5.31A is not acceptable limit or criteria, it is an objective		X …overall limits or criteria related to…		
France	5	3.21 (c)	are presented in para~~s 5.31 and~~ 5.31A of SSR-2/1 (Rev.1) [1]. Member States may choose to apply more restrictive acceptable limits or criteria for design extension conditions without significant fuel degradation. For example, some Member States choose to apply identical or similar overall acceptable…	3.21c : • 5.31 of SSR-2/1 is for practical elimination (even if it is written in a general chapter for DEC) thus, it is not relevant to quote it in chapter 3 of DS 508, • 5.31A is not really related to limit or criteria, the use of "overall" as at the first part of 3.21c is adequate and shall be maintained.	X	…are presented in para 5.31A of SSR-2/1 (Rev.1) [1]. Member States may choose to apply more restrictive acceptable limits or criteria for design extension conditions without significant fuel degradation. For example, some Member States choose to apply identical or similar overall limits or criteria…		To be consistent with previous modification.

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Germany	7	3.22	If it is possible to use available safety systems to respond to design extension conditions without significant fuel degradation, deterministic safety analysis is still required to demonstrate their effectiveness: see Requirement 42 of SSR-2/1 (Rev. 1) [1]. The deterministic safety analysis may use less conservative methods and assumptions than for design basis accidents (see 3.21). Nevertheless, there should still be adequate confidence in the results of the deterministic safety analysis and the safety margins to avoid cliff edge effects should be demonstrated to be adequate (see paras 7.45 and 7.54 to 7.55 of SSG-2 (Rev. 1) [9])	The wording "deterministic safety analysis" is occurring in the very last version of this document. We cannot re-trace the comments of SSC members to do so. Requirement 42 "Safety analysis of the plant design" of SSR-2/1 (Rev.1) states that both, deterministic analysis and probabilistic analysis shall be applied, which was clear depicted in the previous text of para. 3.22. Are there any special reasons to insert "deterministic" in this place? Does it imply that probabilistic methods are not required for this case? Otherwise, we think it is a good combination: the first sentence is about both analysis, the second and the third one – about peculiarities of deterministic analysis for this special case. Please put para 3.22 in line with SSR-2/1 (Rev.1) and delete "deterministic" here.	X			
Finland	3.	3.33	First sentence: "In order to avoid the threat to the containment integrity due to overpressurization, the pressure inside the containment should be controlled." Footnote can be deleted, as well.	The increasing leak rate from the containment is not usually the main reason to control the containment pressure, especially in severe accidents, but rather to maintain the pressure below the design pressure and to avoid the containment failure.	X			

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UK	3	3.33	Original wording – "In particular, as the actual leak rate of the reactor containment increases by a higher the reactor containment pressure is, this pressure should be controlled." Change to read "In particular, as the leak rate of the reactor containment is a function of the reactor containment pressure, the pressure should be controlled to minimise the leakage.".	doesn't make sense as worded.		X In order to avoid the threat to the containment integrity due to overpressurization, the pressure inside the containment should be controlled.		Text change proposed by Finland comment 3 (see above).
France	6	3.33	~~The source term inside the containment in design extension conditions with core melting is such that the potential radioactive releases from any direct leakage to the environment have to be avoided or minimised by providing a safety limit leak rate for the reactor containment, as stated in para 4.100 of SSG-53 [6]. Additional potential paths of leakage of radioactive releases (e.g. containment penetrations) may be identified and measures need to be taken to avoid and reduce the impact of those radioactive releases to the environment (e.g. collect and filter such leakages). Considering the reactor containment structural integrity is ensured, radioactive releases from a direct leakage are a consequence of the leak rate originated from the reactor containment structure depending on the load conditions during accident conditions (see para. 4.28 of SSG-53 [6]).~~	disagree: 4.100 of SSG-53 does not say that and a safety limit rate of the containment does not aim at avoiding direct leakage Other leakage path shall be firstly prevented before filtered If the containment integrity is ensured, a direct leakage does not originate from the load. 4.28 of SSG-53 does not say that Please delete the full article	X			First part of the text is deleted. The rest is modified according to other comments.

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France	7	3.33	… containment pressure[21] is	Delete footnote	X			
ENISS	2	3.33	“The source term inside the containment in design extension conditions with core melting is such that the potential radioactive releases ~~from any direct leakage~~ to the environment have to be avoided or minimised by providing a safety limit leak rate for the reactor containment, as stated in para 4.100 of SSG-53 [6]. ~~Additional~~ Potential paths of leakage of radioactive releases (e.g. containment penetrations) may be identified and measures need to be taken to avoid and reduce the impact of those radioactive releases to the environment (e.g. collect and filter such leakages). Considering the reactor containment structural integrity is ensured, radioactive releases ~~from a direct leakage~~ are a consequence of the leak rate originated from the reactor containment structure depending on the load conditions during accident conditions (see para. 4.28 of SSG-53 [6])	As per par 2.7 of SSG53 recalled below and para 4.100 defining the safety limit leak rate as “the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions”, there is no “additional leakage”, all potential paths of leakage being part of the safety limit leak rate, otherwise radiological consequences calculations are under-evaluated. Some leakage paths may be through adjacent building and their final contribution to external radiological consequences reduced through collection and filtering, but they are part of the overall leak rate. In the same way, it’s not only the direct but also the additional paths of leakage (penetrations…) that are dependent on the containment pressure. Practically the leakage is a function of the pressure difference between the inside of the containment and the area outside of it where the leak is happening, this area could be a pressurised area. See also SSG 53 : 2.7. The leaktightness of the containment is essential to confine radioactive material and to minimize radioactive releases. Leaktightness is generally characterized by specified maximum leak rates (overall leak rate and specific leak rates for containment			X	Text deleted as proposed by France comment 6. (see above)

MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
				penetrations, air locks, hatches and containment isolation valves) that are not expected to be exceeded under accident conditions.				
ENISS	3	3.33	"This may be achieved by provision ensuring and maintaining ~~of~~ adequate cooling of the reactor containment atmosphere during the design extension conditions with core melting or by a filtered reactor containment venting system allowing to reduce the containment pressure ~~radioactive releases~~ or other design features or alternative measures. ~~Therefore~~, The ultimate consequences of filtered and unfiltered direct leakage of radioactive releases from the reactor containment in design extension conditions with core melting should remain below the design target defined as per recommendations of SSG 53 para 2.7 and 2.11 and assessed as per recommendation of SSG 53 para 11.7 ~~safety limit leak rate for the reactor containment~~ to allow sufficient time for implementation of off-site protective actions. At any ~~Beyond this~~ time, radiological releases should ~~might exceed the safety limit leak rate for the reactor containment but should still~~ be well below the radioactive releases considered as an early or large radioactive release	This paragraph still needs clarification: A FCV (filtered containment venting) may ultimately reduce the overall radiological consequences calculated over a certain period of time, but the first mission of a FCV is not to reduce the releases, but to control the containment pressure by filtration of an intended release. How could you reduce leakages while using a FCV creates some? The "therefore" is creating a confusing link. The conclusion on "unfiltered leakage" is misleading. The radiological consequences to the people and the environment are assessed from the summation of "unfiltered + if any, the controlled and filtered leakages". As explained in SSG 53 § 4.100 (the leak rate assumed in the assessment of possible radioactive releases arising from accident conditions), the "safety limit leak rate for the reactor containment" is not a target to be reached in deterministic safety analyses, but an assumption for radio-logical consequences assessment. The impact on people and the environment is not measured through a leak rate, but through radiological consequences calculations. (The impact		X This may be achieved by ensuring and maintaining adequate cooling of the reactor containment atmosphere during the design extension conditions with core melting or by a filtered reactor containment venting system allowing to reduce the containment pressure or other design features or alternative measures, as mentioned in para 11.8 of NS G 1.13 [13]. The ultimate consequences of filtered and unfiltered direct leakage of radioactive releases from the reactor containment in design extension conditions with core melting should remain below the design target defined as per recommendations of para 2.7 of SSG-53 [6] and para 2.10 of NS G 1.13 [13] and assessed as per		To be make reference to correct paras in references.

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				of a flow of water on the environment is the level of water ultimately reached not just a question of limiting a flow rate). As stated in NS-G-1.13 Para 2.10 and 2.11 : “2.7. To ensure that a design both reduces doses to levels that are as low as reasonably achievable and represents best practice, design targets should be set for the individual dose and collective dose to workers and for the individual dose to those members of the public who will receive the greatest doses. 2.10. The adequacy of the design provisions for the protection of the site personnel and public under postulated accident conditions should be judged by means of the comparison of calculated doses with the specified dose criteria that constitute the design targets for accidents. In general, the higher the probability of the accident condition, 2.11 […] For severe accidents, the regulatory body may specify a risk criterion or a criterion associated with specified releases of radioactive substances. 11.1. The possible consequences of design basis accidents and severe accidents should be determined to demonstrate compliance with design targets. 11.7. For severe accident scenarios, specific analysis should be performed to demonstrate compliance with national		recommendation of para 11.7 of NS G 1.13 [13] to allow sufficient time for implementation of off-site protective actions. At any time, radiological releases should be well below the radioactive releases considered as an early radioactive release or a large radioactive release.		

MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
				regulatory requirements concerning both the short term and the long-term consequences of an accident" Is the last sentence consistent with SF-1 principle 5 (Protection must be optimized to provide the highest level of safety that can reasonably be achieved.)?				
Germany	8	3.33	The source term inside the containment in design extension conditions with core melting is such that the potential radioactive releases from any direct leakage19 to the environment have to be avoided or minimised by providing a safety limit leak rate for the reactor containment, as stated in para 4.100 of SSG-53 [6]. Additional potential paths of leakage of radioactive releases (e.g. containment penetrations) may be identified and measures need to be taken to avoid and reduce the impact of those radioactive releases to the environment (e.g. collect and filter such leakages). ~~Considering the reactor containment structural integrity is ensured, r~~Radioactive releases from a direct leakage are a consequence of the leak rate originated from the reactor containment structure depending on the load conditions during accident conditions (see para. 4.28 of SSG-53 [6]). In particular, as the actual leak rate of the reactor containment increases by a higher ~~the~~ reactor containment pressure[20] ~~is~~, this pressure should be controlled. This may be achieved by	We made few changes in text (deleting the redundant parts) to make the text readability.		X In order to avoid the threat to the containment integrity due to overpressurization the pressure inside the containment should be controlled. This may be achieved by ensuring and maintaining adequate cooling of the reactor containment atmosphere during the design extension conditions with core melting or by a filtered reactor containment venting system allowing to reduce the containment pressure or other design features or alternative measures, as mentioned in para 11.8 of NS G 1.13 [13]. The ultimate consequences of filtered and unfiltered direct leakage of radioactive releases from		Text modified consider other comments (see above).

MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			ensuring and maintaining adequate cooling of the reactor containment atmosphere during the design extension conditions with core melting or by a filtered reactor containment venting system allowing to reduce the radioactive releases. Therefore, unfiltered direct leakage of radioactive releases from the reactor containment in design extension conditions with core melting should remain below the safety limit leak rate for the reactor containment to allow sufficient time for implementation of off-site protective actions. Beyond this time, releases might exceed the safety limit leak rate for the reactor containment but should still be well below the radioactive releases considered as a large radioactive release.			the reactor containment in design extension conditions with core melting should remain below the design target defined as per recommendations of para 2.7 of SSG-53 [6] and para 2.10 of NS G 1.13 [13] and assessed as per recommendation of para 11.7 of NS G 1.13 [13] to allow sufficient time for implementation of off-site protective actions. At any time, radiological releases should be well below the radioactive releases considered as an early radioactive release or a large radioactive release		
Germany	9	Footnote 19 (20?) Page 14	At some point the pressure inside of the reactor containment may be so high that the reactor containment may start to fail. This is a cliff edge effect to be avoided.	What was the reason to delete the phrase "This is a cliff edge effect to be avoided"? A 'cliff edge effect' is defined in the IAEA Safety Glossary [3] as "An instance of severely abnormal conditions caused by an abrupt transition from one status of a facility to another following a small deviation in a parameter or a small variation in an input value." The term 'parameter' in this definition can be interpreted in a broad sense as any plant			X	Text deleted to consider modification of the text in para 3.33 (see above).

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				physical variable, design aspect, equipment condition or magnitude of a hazard that can influence equipment or plant performance. Specific examples of cliff edge effects are rare, we suggest to leave such an example here, in this footnote.				
Finland	4.	3.38	"… level that is as low as reasonably practicable, also considering…"	Change "achievable" to "practicable" to be consistent with the terminology.			X	The radiological risk has to be reduced to the level as low as reasonably achievable but the levels of DiD have to be independent as far as is practicable. See SSR-2/1 (Rev.1).
Finland	5.	3.49	"… they effectively reduce challenges to safety systems…"	Remove "the number of" since it is not needed, and furthermore it is somewhat misleading, as severity may be of more importance than the number.	X			
Germany	10	3.49	The reliability of structures, systems and components for controlling anticipated operational occurrences should be such that they effectively reduce the number of challenges to safety systems and contribute to preventing the occurrence of ~~design basis accidents~~ accident conditions.	There seems to be a technical mistake here – the suggestion of Japan has been accepted already during Step 11 review. We support the Japans comment and suggest the wording "preventing the occurrence of accident conditions" instead of "preventing the occurrence of design basis accidents". The reason: Controlling of anticipated operational occurrences will contribute to	X			

MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
				preventing the occurrence not only of “design basic accidents”, but “design extension conditions” as well, which all together are encompassed in “accident conditions”.				
Finland	6.	3.50	“…should be such that the core damage frequency…” and add: “Design extension conditions without significant fuel degradation should be postulated (see paras 3.39 to 3.44 of SSG 2 (Rev. 1) [9]) and analyzed considering applicable analysis rules (see paras 7.45-7.55 of SSG-2 (Rev. 1) [9]) as appropriate to achieve the safety goals.”	Remove “the collective contribution to”, as the contribution would assume only part of the CDF, but here the goal is to achieve an overall CDF below a set value. Please add part of the para. 3.52 to 3.50 e.g. para3.52. “Design extension conditions without significant fuel degradation should be postulated (see paras 3.39 to 3.44 of SSG 2 (Rev. 1) [9]) and analyzed considering applicable analysis rules (see paras 7.45-7.55 of SSG-2 (Rev. 1) [9]) as appropriate to achieve the safety goals.”	X			
Finland	7.	3.52	Please delete 3.52. See also comment 3.50	This seems to say the same as the first part of 3.50, but with different wording, which may cause confusion. It is suggested to check, if there is a real need to introduce such a text twice, and what is the essential message of the both. Remove part of para. 3.52. “Design extension conditions without significant fuel degradation should be postulated (see paras 3.39 to 3.44 of SSG 2 (Rev. 1) [9]) and analyzed considering applicable analysis rules (see paras 7.45-7.55 of SSG-2 (Rev. 1) [9]) as appropriate to achieve the safety goals.” into para. 3.50.	X			

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MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
Finland	8.	3.53	"…postulated core melt sequence (see para. 3.28). …"	Reference to 3.28 should be added, as otherwise the link to the selection process remains unclear.	X			
Finland	9.	3.53	Replace the second sentence with "As there may be large uncertainties associated with the analyses of core melt accidents, these should be taken into account when evaluating the reliability of the safety features."	The original second sentence starting with "However," gives a too pessimistic message. It could be read that the reliability would not be good in any case.	X			
Finland	10.	3.58	Suggestion to remove "Because of these factors," from the beginning and start with "Full…"	This para does not explain,why full independence cannot be achieved, as implied when starting with "Because of…". There are other reasons, why full independence cannot be achieved, or it is not reasonable to try that, e.g. control rods and the containment structure are important in many levels of DiD. The essential point is that the DiD principle can be followed, i.e. the means to manage the situation remain, although one of the levels fail, not on what level of DiD some system is allocated to. This is explained already in 3.59.	X			
Germany	11	3.58	~~Because of these factors, full independence of the levels of defence in depth cannot may be difficult to achieved.~~ Because of these factors, full independence of the levels of defence in depth may be difficult to achieved. The design of a nuclear power plant should consider all potential causes of dependencies and an approach should be implemented to remove them to the extent reasonably practicable. Robust	It seems to be a technical mistake here. Draft Version of DS508, published for Step 11 review (before 53.NUSSC) contains the following wording in para. 3.58: 3.58 Because of these factors, full independence of the levels of defence in depth **cannot be achieved**. The design of a nuclear			X	The text from the proposal of UK comment 18, before the 53rd NUSSC meeting was modified to acknowledge that there are several factors and constraints, such as internal hazards and external hazards, as well as the actual design of some key SSCs, such as the containment, which cannot

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			independence should be implemented among systems whose simultaneous failure would result in conditions having harmful effects for people or the environment.	power plant should consider all potential causes of dependencies and an approach should be implemented to remove them to the extent reasonably practicable. Robust independence should be implemented among systems whose simultaneous failure would result in conditions having harmful effects for people or the environment. During the review process UKs comment 18 submitted to current para, and the resolved text (published after Step 11 review) contains the following wording in para. 3.58: 3.58 Because of these factors, full independence of the levels of defence in depth **may be difficult to achieved.** The design of a nuclear power plant should consider all potential causes of dependencies and an approach should be implemented to remove them to the extent reasonably practicable. Robust independence should be implemented among systems whose simultaneous failure would result in conditions having harmful effects for people or the environment. We cannot trace the reasons, for which a new text of 3.58 (**may be difficult to achieved**) should be converted into the				allow to reach full independence as these SSCs are required at different levels of DiD. The idea of this text was previously presented, discussed and accepted in the version of DS508 step 9 after the CS conducted in September 2021 before the NUSSC 53rd meeting. The recommendations that follow are in accordance with this fact.

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				former one (**cannot be achieved**). Please check this matter.				
France	8	3.59	…parts of them for executing safety related functions for different plant states should be avoided	Check definition of safety related because this expression is not relevant here	X			
Finland	11.	3.62	"… A postulated initiating event…"	Change "postulating" to "postulated".	X			
France	9	3.62-3.66	Would it be deleted or not?	From previous discussion.			X	It was decided during the NUSSC 53rd meeting that this section remains.
Finland	12.	3.62	Change the last sentence to "The adequacy of the independence should also be assessed by probabilistic analyses."	Delete "that is achieved for each level of defence in depth" from the sentence. The independence of each level is not usually evaluated by probabilistic approach, but rather the adequacy of the overall outcome to avoid core melting.		X The adequacy of independence between levels of defence in depth should also be assessed by probabilistic analyses.		The independence between levels could be assessed by probabilistic safety assessment, for instance by the analysis of MCS of relevant accident sequences to avoid core melting.
Finland	13.	3.66	The last sentence "In particular, a common cause failure should not affect at the same time the safety functions performed by the safety systems or some safety features for design extension conditions without significant fuel degradation and the safety functions of the necessary safety features for design extension conditions for core melting." Should be removed.	This is related to the design not to the assessment. This is generally said elsewhere (e.g. in 3.51).		X In particular, the assessment should be conducted to ensure that a common cause failure will not affect at the same time the safety functions performed by the safety systems…		The assessment of CCF affecting different levels of DiD is also part of the safety assessment not only of the design safety. The text has been modified to better reflect this recommendation which is different to the recommendation in para 3.51.
Finland	14.	4.2	"This requirement is essentially introduced also in SSR-2/1 para 5.31."	The requirement is not exactly the same, and it has a slight difference.	X			
Finland	15.	4.5	"…As a result of the proper implementation of the first, second,	Add "proper" and "for most cases". Only implementation of such levels	X			

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			third and fourth levels of defence in depth, the likelihood of an off-site radioactive release that could potentially result from an accident will be very low for most cases. …”	does not necessarily guarantee the efficiency of these measures. Furthermore, it is not always possible to add some measures to one level to avoid the escalation to the next one.				
Finland	16.	4.7	Remove the whole para.	The message of the para is confusing and should thus be removed. Sometimes mitigation is seen as an essential part of the practical elimination when considering technical means related to the plant design and operation.		4.7 Therefore, as mentioned in para. 4.4, the concept of practical elimination should be applied only in relation to plant event sequences that could lead to an early radioactive release or a large radioactive release, for which reasonably practicable technical means for their mitigation cannot be implemented. For other accidents that might lead to a radioactive release not considered for the application of the practical elimination, the technical means should be considered in the design for the mitigation of such accident consequences at the plant, but this would not constitute the application of the concept of practical elimination.	X	Text modified to better provide the recommendation related to the different between the application of the practical elimination concept and those accidents that need to be mitigated by the design.
UK	4	4.7	Original wording – “Therefore, as mentioned in para. 4.4, the concept of	Prior to NUSSC53 (refer to ONR email 13/6/22) , ONR suggested that this		X		Text modified to better provide the recommendation related to

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			practical elimination should be applied only in relation to plant event sequences that could lead to an early radioactive release or a large radioactive release, for which reasonably practicable technical means for their mitigation cannot be implemented. Otherwise, the technical means should be considered in the design for the mitigation of the accident consequences at the plant, but this would not constitute the application of the concept of practical elimination." Proposed change – delete whole of paragraph 4.7 or get some clarification from ENISS on the intended meaning.	paragraph should be deleted. The first part of this paragraph has already been stated earlier in Section 4. It is not clear what the final sentence is trying to say – it seems to be at odds with paragraph 4.6 which states that application of practical elimination may result in the identification of additional provisions – these would need to 'reasonably practicable technical means'. The UK's preference would still be to delete this paragraph as it doesn't add value and is potentially confusing given the other text. In the side-discussions at NUSCC 53, we have a recollection that the ENISS representative indicated to the UK that this paragraph was very important to them. We can respect that, but as currently written, after several re-reads, we just do not understand the important points that are trying to be made. Perhaps with an editorial change (in particular the final sentence), it will become clearer. The use of the word "Otherwise" might be part of our issue. Without understanding the full meaning, it is difficult to propose alternative words. We think it is trying to say that accidents with consequences that do not lead to large or early releases do still need to be considered in the design, but this is		4.7 Therefore, as mentioned in para. 4.4, the concept of practical elimination should be applied only in relation to plant event sequences that could lead to an early radioactive release or a large radioactive release, for which reasonably practicable technical means for their mitigation cannot be implemented. For other accidents that might lead to a radioactive release not considered for the application of the practical elimination, the technical means should be considered in the design for the mitigation of such accident consequences at the plant, but this would not constitute the application of the concept of practical elimination.		the different between the application of the practical elimination concept and those accidents that need to be mitigated by the design.

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				achieved through the application of the concept of defence in depth (already discussed at length in DS508) and not directly through the application of practical elimination concepts.				
Finland	17.	4.8	Add footnotes in "Independent of the design or specific definitions of the phrases, early radioactive releases or large radioactive releases are those which will challenge defence in depth Level 5 provisions." to explain early release and large release.	The definitions from Safety Glossary should be introduced or referred to here. "early release of radioactive material: A release of radioactive material for which off- site protective actions are necessary but are unlikely to be fully effective in due time." "large release of radioactive material: A release of radioactive material for which off-site protective actions that are limited in terms of times and areas of application are insufficient for protecting people and the environment." Otherwise, it remains unclear if there isn't anything in the IAEA standards on these.	X			Remark: Recommendations related about further defining the early radioactive release frequency and large early release frequency are defined in DS528 (revision of Level 2 PSA SG (SSG-4)) to be presented for the NUSSC 55th meeting.
Germany	12	4.8	SSR-2/1 (Rev. 1) [1] does not provide quantitative acceptance limits or criteria for the radiological consequences of accident conditions, nor for the magnitude of what is to be considered an early radioactive release or a large radioactive release. Independent of the design or specific definitions of the phrases, early radioactive releases or large radioactive releases are those which will ~~could~~ challenge defence in depth Level 5 provisions. In some States an	Do we understand correctly that the statement "However, the justification that a plant event sequence has been practically eliminated should rely primarily on a deterministic evaluation of the robustness and independence of design safety provisions and should not solely relied on the compliance with such probabilistic criteria, but supported by the results of probabilistic safety assessments" has been deleted because of UK comment during the 53rd NUSSC meeting and France				Deleted since it was a repetition of para 4.35. Added again here for reconsideration.

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			early radioactive release is defined for a specific site considering restrictions on implementing off-site protective actions in a timely manner. In some States, acceptable limits on radioactive releases for purposes of radiation protection, and probabilistic criteria or target values for the purpose of demonstrating a low frequency of a core damage accident, have been established, consistent with regulatory requirements or objectives. However, the justification that a plant event sequence has been practically eliminated should rely primarily on a deterministic evaluation of the robustness and independence of design safety provisions and should not solely relied on the compliance with such probabilistic criteria, but supported by the results of probabilistic safety assessments.	comment after the 53rd NUSSC meeting, is this so? Can you please explain the reasons for changing, or deleting the phrase, starting from "however"? The statement that it is not possible to prove practical elimination using only probabilistic arguments is very important in this guide. As stated in TEDOC-1791 (Section 7.1), "there is a quite wide consensus on the view that the 'practical elimination', even involving probabilistic considerations, always needs to be based on solid design provisions and supported by deterministic assessment and engineering judgement." This is also indicated in para. 4.35 of this draft, however not in sufficient clarity. Our opinion is that the current statement should be mentioned in para 4.8, especially because of the connection with the acceptance criteria. In this way, it can be made clear that proof of fulfillment of quantitative acceptance criteria is not sufficient to justify practical elimination. We would like to ask you kindly to integrate the above issue back into para 4.8.				
Finland	18.	4.11	"… in the fuel or within the reactor coolant system…"	Change "by" to "within".	X			

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Russian Federation	1	4.13 Footnote 26	"In the case when a spent fuel pool is located inside the containment, the containment provides an additional barrier that prevents direct release of radioactive substances into the environment. In this case, any significant fuel degradation in the spent fuel pool **does not directly lead to** a large radioactive release into the environment. Only significant fuel degradation in the spent fuel pool followed by subsequent penetration of the base of the spent fuel pool and the basement of the containment can lead to a large radioactive release into the environment. In this case, since additional protective technical means could be practically unrealizable in design, plant event sequences resulting on damage of the containment basement has to be considered for practical elimination".	The footnote focuses only on the NPPs with a spent fuel pool located outside of the containment. To make this footnote more universal in terms of accounting for the existing NPP design solutions on the location of the spent fuel pool, the Russian Federation proposes to reflect the following information in this footnote (in addition to the current text of the footnote related to outside spent fuel pool):		In the case when a spent fuel pool is located inside the containment, the containment provides an additional barrier that prevents direct release of radioactive substances into the environment. In this case, any significant fuel degradation in the spent fuel pool does not directly lead to a large radioactive release into the environment. Only significant fuel degradation in the spent fuel pool followed by subsequent penetration of the base of the spent fuel pool and the basement of the containment can lead to a large radioactive release into the environment. In this case, since additional protective technical means could be practically unrealizable in design, plant event sequences resulting on significant fuel degradation in the spent fuel pool followed by subsequent penetration of the base of the spent fuel pool and damage of the containment basement has to be considered for practical elimination.		The added text aims at reflecting spent fuel pools located inside a containment building, such as the reactor containment building. However, the sequence to be considered for practical elimination starts with the significant fuel degradation of the fuel stored in the spent fuel pool, since there are no additional safety features in the design to manage and control significant fuel degradation of fuel stored in the spent fuel pool inside or outside of the containment, such as the Ex-Vessel Corium Cooling.
Finland	19.	4.15	Add text "Also, some bypass sequences in 4.13 (d) may involve adequate	As mitigation is addressed to some of the sequences, it would be worthwhile	X			

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			natural retention of radioactive substances to achieve the safety goal.”	to mention this aspect, as well.				
Germany	13	4.15	~~Other criteria for grouping are also possible. The consequences of the accidents in para. 4.13(c)(i) and 4.13(c)(ii) could in fact be mitigated by the implementation of reasonable technical means. In such cases, for scenarios not retained within the scope of consideration for practical elimination, evidence of the effectiveness and an appropriate reliability of the mitigation should be provided.~~ ~~To facilitate the grouping proposed, each type of plant event sequence should be analysed to identify the associated combination of failures or associated physical phenomena that are specific to the plant design, and which have the potential to lead to a loss of the confinement function.~~ Other criteria for grouping are also possible. To facilitate the grouping proposed, each type of plant event sequence should be analysed to identify the associated combination of failures or associated physical phenomena that are specific to the plant design, and which have the potential to lead to a radioactive release greater than the maximum radioactive release allowed in accordance with para 5.31A of SSR-2/1 (Rev.1) [1].	It seems to be a technical mistake here. Draft Version of DS508, published for Step 11 review (before 53.NUSSC) contains the following wording in para. 4.15: 4.15 Other criteria for grouping are also possible. The consequences of the accidents in para. 4.14(c)(i) and 4.14(c)(ii) could in fact be mitigated by the implementation of reasonable technical means. In such cases, for scenarios not retained within the scope of consideration for practical elimination, evidence of the effectiveness and an appropriate reliability of the mitigation should be provided. To facilitate the grouping proposed, each type of plant event sequence should be analysed to identify the associated combination of failures or associated physical phenomena that are specific to the plant design, and which have the potential to lead to a loss of the confinement function. During the review process a number of comments have been submitted from SSC Members, so the resolved text (published after Step 11 review)			X	The text was restored to its previous proposal considering France comment 9 step 11 (see pdf file “DS508 - Table of SSCs comments resolution” from 08/06/2022). In addition, it is better to refer to the loss of the confinement function instead to the maximum radioactive release that could be considered with regard to para 5.31A of SSR-2/1 (Rev.1).

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				contains the following wording in para. 4.15: 4.15. Other criteria for grouping are also possible. To facilitate the grouping proposed, each type of plant event sequence should be analysed to identify the associated combination of failures or associated physical phenomena that are specific to the plant design, and which have the potential to lead to a radioactive release greater than the maximum radioactive release allowed in accordance with para 5.31A of SSR-2/1 (Rev.1) [1]. We cannot trace the reasons, why the new text of 4.15 has been converted back into the old, previous one.				
UK	5	4.19	Original wording - “No need to conduct on-site actions of use off-site personnel or equipment”. Change to - “minimisation of on-site actions and the use of off-site personnel or equipment”.	The sentence does not flow from the introduction before the list, ie “…..should consider the following aspects:”		X … (g) Reduce the No need to conduct on-site actions or use off-site personnel or equipment		As proposed by ENISS.
ENISS	4	4.19g	“Reduce the ~~No~~ need to conduct on-site actions or use off-site personnel or equipment”	There are 2 aspects in bullet g of para 4.19: “(g) No need to conduct on-site actions or use off-site personnel or equipment” That are: (g1) No need to conduct on-site actions	X			

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				(g2) No need to use off-site personnel or equipment" Although g2 is already challenging by some member states g1 means "no possible actions from operators either from the control room or locally". This is a strong not acceptable recommendations, which is not inline for some existing advanced designs for the provisions described in the Annexe I. This will also be in contradiction with the para 6.2.c and para 7 of the WENRA paper on practical elimination. Furthermore 4.19 g1 is not consistent with para 4.23. 4.23 Safety provisions for demonstrating practical elimination of some severe accident conditions could include first the need of design provisions as well as operational provisions, and as such they could involve the performance of operator actions (e.g. the opening of primary circuit depressurization valves to prevent high-pressure core melt conditions).				
Germany	14	4.20	The identification of safety provisions necessitates a comprehensive analysis of the physical phenomena involved, from the deterministic, probabilistic and engineering judgement perspectives, and it might be necessary to further refine the identification of event sequences performed in	It seems to be a technical mistake here: As a reaction to Canadas comment 36 before the 53. NUSSC meeting the phrase "from the deterministic, probabilistic and engineering judgement perspectives" has been added. Was there any reason to delete this formulation? We cannot trace a	X			

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			accordance with the approaches described in para. 4.16.	request to delete this part. We suggest to restore the text.				
Finland	20.	4.23	Replace "Requiring operator actions should be minimized and, when unavoidable, …" with "The amount of operator actions should be limited, and when included, …"	It is not feasible to aim at minimizing the operator actions, but rather limiting their amount.		X Modified as text proposed by UK comment 6.		See text proposed by UK comment 6. (below)
UK	6	4.23	Original wording – "Safety provisions for demonstrating practical elimination of some severe accident conditions could include first the need of design provisions as well as operational provisions , and as such they could involve the performance of operator actions (e.g. the opening of primary circuit depressurization valves to prevent high-pressure core melt conditions). Requiring operator actions should be minimized and, when unavoidable, a human factor assessment should be part of the justification supporting any claim for high reliability of operator actions. The human factor assessment should address the following: (a) The availability of information given to operating personnel to perform the actions from the control room or locally, and the quality of the procedures or guidelines to implement the actions, and the training of the required operating personnel;" Change to - "Safety provisions for demonstrating practical elimination of	minor typos & readability. (b) & (c) are OK as is.				

MS	Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/rejection
			some severe accident conditions could include the need for design provisions as well as operational provisions, and as such they could involve the performance of operator actions (e.g. the opening of primary circuit depressurization valves to prevent high-pressure core melt conditions). Requirements for operator actions should be minimized and, when unavoidable, a human factor assessment should be part of the justification supporting any claim for high reliability of operator actions. The human factor assessment should address the following: (a) The availability of information given to operating personnel to perform the actions from the control room or locally, the quality of the procedures or guidelines to implement the actions and the training of the required operating personnel;"					
Finland	21.	4.27	"… safety provisions included in the practical elimination should be demonstrated…"	It is not important who has identified these provisions		X 4.27 The overall effectiveness of the safety provisions identified and included ~~by the designer~~ to demonstrate practical elimination should be demonstrated through a safety assessment that includes engineering judgement, deterministic		The safety provisions for the demonstration of the PE concept should be first identified and later included.

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						analyses and probabilistic assessments.		
Germany	15	4.34	In practice, the demonstration of physical impossibility is limited to very specific cases <u>(see Annex I).</u> Demonstration of physical impossibility cannot rely on measures that involve active components or operator actions. *Please integrate the second mentioned example concerning the practical elimination of post-accident combustible gas detonations that can harm the integrity of the containment in Annex I as well. It might be added to para I-24.*	It is a good idea to remove the examples in Annex I. However, we are missing a reference to Annex I in this place and suggest to add. The second example, concerning the practical elimination of post-accident combustible gas detonations that can harm the integrity of the containment, is missing in Annex I. It could be added to para I-24 to clarify that the justifying of this practical elimination is possible by the demonstration of physical impossibility due to a limited amount of material that could generate combustible gas during a severe accident.	X	X I.24… This assessment also includes the consideration of first the appropriate selection of materials allowing a limited amount of hydrogen generation during severe accident and second the hydrogen propagation and mixing inside the containment.		
Finland	22.	4.35	"… possible implementation of additional reasonably practicable safety provisions…"	Replace "reasonable" with "reasonably practicable".	X			
Finland	23.	4.41	If the plant event sequence to be practically eliminated is the result of a single initiating event, such as the failure of a large pressure-retaining component1 in normal operation, the demonstration of practical elimination should rely on the substantiation that a high level of quality is achieved at all	Please correct the consequence in last sentence. It is unclear how reactivity accident is connected to the vessel breach, and therefore this "and the consequential event (i.e. uncontrolled reactivity accident)" is confusing. Please correct the consequence in line with para. 4.13	X			

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			stages of the lifetime of the component, i.e. its design, manufacture, implementation, commissioning and operation (including periodic testing and in-service surveillance, if any) so as to prevent the occurrence and propagation of any defect liable to cause the failure of the component. Hence, both the occurrence of the single initiating event (e.g. failure of a large pressure-retaining component) and the consequential event (i.e. uncontrolled reactivity accident lead to prompt reactor core damage and consequent early containment failure) should be considered for practical elimination.	a) lead to prompt reactor core damage and consequent early containment failure.				
Finland	24.	4.42	"… confinement function is degraded in such an extent that adequate retention of radioactive substances is not possible before core melt…"	Minor degradation does not necessarily lead to unacceptable releases.	X			
Finland	25.	5.3	"To provide additional resilience against event sequences exceeding those considered as a basis for design or design, such as levels of external natural hazards, several requirements…"	In the first part change "considered" to "considered as a basis for design". An event and its severity are selected as a basis for design through some process. In addition to this a design margin is set, and this becomes a new design basis for e.g. flood protection. Therefore, there is no need for considering events more severe than this new design basis if it includes adequate margins already. Otherwise, this becomes a never-ending process to take into account more and more		X … those considered as the basis for the design, such as levels of external natural hazards exceeding those considered in the design basis derived from the hazard evaluation for the site, …		Deleting the text "…exceeding those considered in the design basis …leads to incomplete explanation. It is better to keep the text after adding the previous proposal.

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				severe events. Consequentially "exceeding those considered in the design basis" can be removed from the original sentence.				
France	10	5.3	in the design basis derived from the hazard evaluation for the site, several requirements	To be consistent with SSR-2/1	X			
Finland	26.	Paras 5.5, 5.6, 5.7, 5.8, 5.8 (a), 5.10, 5.11	"…exceeding the levels considered as a basis for the design…"	See above.	X			
Finland	27.	5.6	To be moved to a more general part or modify the title of Chapter 5.	This is a god para, but it does not have connection to non-permanent equipment.		X … The behaviour of structures, systems and components to loading parameters resulting from these levels should be assessed with regard to potential use of non-permanent equipment (e.g. coping time for deployment)		Text added to be in relation to non-permanent equipment.
France	11	5.6	…for design derived from the hazard evaluation for the site should	To be consistent with SSR-2/1	X			
France	12	5.6	…by the addition of a relevant <u>margin</u> .	It is not a margin : severity?			X	It is a margin that is added.
France	13	5.7	the levels considered for the design design derived from the hazard evaluation for the site as follows:	To be consistent with SSR-2/1	X			
Finland	28.	5.8	To be moved to a more general part or modify the title of Chapter 5.	This is good text, but it misses to specify its importance to non-permanent equipment.			X	The non-permanent equipment is mentioned in the brackets.

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France	14	5.8	design derived from the hazard evaluation for the site , the evaluation should	To be consistent with SSR-2/1	X			
Finland	29.	5.10	"… level of natural hazard exceeding…"	Remove either "natural" or "external".		X level of ~~natural~~ external hazard exceeding those considered as the basis for the design derived from the hazard evaluation for the site (such natural external hazards as earthquake).		The example of natural external hazard is mentioned.
France	15	5.10	exceeding those considered for the design derived from the hazard evaluation for the site .	To be consistent with SSR-2/1	X			
Finland	30.	5.10, 5.11, 5.12	To be moved to a more general part or modify the title of Chapter 5.	These are not related to non-permanent equipment only.		X Text modified to consider non-permanent equipment.		
Germany	16	Definition	Practical elimination Plant event sequences that could lead to an early radioactive release or a large radioactive release are either physically impossible or are demonstrated, with a high level of confidence, to be extremely unlikely to arise by implementing safety provisions in the form of design and operational features. o The concept of practical elimination is applied in relation to plant event sequences, the consequences of which cannot be mitigated by reasonable	The definition structure emphasizes that only practical elimination due to extreme unlikeliness with a high level of confidence has to be demonstrated, which is not true. Practical elimination due to physical impossibility needs to be demonstrated as well.		X Plant event sequences that could lead to an early radioactive release or a large radioactive release should be demonstrated to be either physically impossible or, with a high level of confidence, extremely unlikely to arise by implementing safety provisions in the form of		The first bullet was not modified since it is considered as correct.

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			practicable means. o Practical elimination is part of a general approach to design safety and is an enhancement of the application of the concept of defence in depth. ***Possible suggestion:*** Plant event sequences that could lead to an early radioactive release or a large radioactive release ~~are~~ should be demonstrated to be either physically impossible or ~~are demonstrated~~, with a high level of confidence, ~~to be~~ extremely unlikely to arise by implementing safety provisions in the form of design and operational features. o The concept of practical elimination is applied in relation to plant event sequences, ~~the~~ whose consequences ~~of which~~ cannot be mitigated by reasonable practicable means. o Practical elimination is part of a general approach to design safety and is an enhancement of the application of the concept of defence in depth			design and operational features. The concept of practical elimination is applied in relation to plant event sequences, the consequences of which cannot be mitigated by reasonable practicable means. Practical elimination is part of a general approach to design safety and is an enhancement of the application of the concept of defence in depth.		