DS 510A Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report (Revision of SSG-20)

COMMENTS BY REVIEWER Reviewer: Page. Country/Organization: Date: 07 June 2019					RESOLUTION			
Comment No.	Country Comment No.	Para/Line No.	Proposed new text	Reason	Accepted	Accepted, but modified as follows	Rejected	Reason for modification/ rejection
1.	Australia 11	General	Remove revision numbers for referenced documents, e.g. Para 1.5 GSR Part 1 (Rev.1)	To prevent the document from becoming obsolete if any referenced document is updated			X	IAEA style of writing references, revision number is mentioned.
2.	Japan 1	General	In para 1.11, it states that this Safety Guide focuses mainly on research reactors of a capacity of up to a few tens of megawatts. It also states that amount of detail required in the safety analysis report is different for small reactor. However, the definition of “small” or “low power” is unclear and should be clarified. The need for water cooling may be one of the essential criteria for categorization of small research reactors. This is the same comment as DS510B, comment #1.			…The amount of detail required for specific research reactors, critical assemblies and sub critical assemblies should be justified and documented using graded approach. Nevertheless, when using a graded approach, all items included in this Safety Guide should be addressed…...		The text is modified as per para 1.8 and 1.9 of SSR-3.
3.	Germany 1	General Whole document	In the course of the revision of SSG-20 an alignment of the structure of the document with the corresponding document for NPPs might be performed	This action increases the comparability of the two documents and simplifies the regulatory use of both documents.			X	It is more appropriate to keep the currently recommended structure of the safety analysis report, as it was in IAEA safety

								standards since 1994. Changes in the format may cause more challenges for Member States.
4.	Germany 2	General Whole document	At the moment, radioactive waste management is a sub-chapter of chapter 12 “OPERATIONAL RADIATION SAFETY”. A dedicated chapter for the management of radioactive waste can include additional aspects.	Waste management comprises more aspects than radioprotection e.g. (pre)-treatment and conditioning of waste which are not covered at the moment by SSG-20			X	It is more appropriate to keep the currently recommended structure of the safety analysis report, as it was in IAEA safety standards since 1994. Changes in the format may cause more challenges for Member States.
5.	Germany 3	General Whole document	A chapter devoted to “human factors” might be introduced.	Human factors engineering, and human-machine interface issues has become more and more important in the operation of research reactors. This should be reflected in the structure of SSG-20. The corresponding NPP document already contains a chapter on this topic.			X	See response to Germany comment #2
6.	Germany 4	General Whole document	Check numbering of footnotes	e.g. page 18. Footnotes 11 and 12 are deleted but a new footnote “13” is introduced.		Accepted and noted		Footnote numbers will be updated automatically at the last step.

7.	Canada 1	General	The overall document, as revised, is significantly clearer and is consistent with safety objectives already contained in SSR 2/1 for NPPs while making allowances for research reactor characteristics. One overarching issue noted throughout the document (specific proposals made to address it in comments below), is that for subcritical nuclear assembly facilities, the document does not fully reflect that there are many sizes and variations of these facilities. The risk profile can vary substantially from very low risk to levels of risk approaching that of a larger research reactor facility if the Keff is high enough while taking into account uncertainties. A number of guidance clauses are written in an 'exclusionary fashion' such as "can be reduced" and "may not be required". Although the intent of this text is to more clearly articulate the use of a graded approach, the reader will automatically assume that less detail or no detail applies in their case and that it is the regulator's role to challenge this. In fact, it is the proponent's role to explain to some degree why an exclusion should be made specific to their case based on risk profile. This does not require significant work by the proponent and provides confidence when SAR information is referenced in public licensing processes. Guidance concerning these issues should always guide a proponent to indicate, using basic risk-informed decision-making approaches why the level of information, if needed, is appropriate for the risk profile of the facility.		X			The approach of developing guidance that covers all research reactors and sub-critical assemblies is the same as it was followed in development of SSR-3. The approach was also described in the DPP of the Safety Guides. The guidance unless specifically mentioned is applicable also to subcritical assemblies with use of a graded approach that commensurate with their potential risk, as described in the Guides. In addition, there will be also a SSG on use of graded approach. Specific comments on some clauses/ paragraphs are provided and resolved as

								described in responses to other comments, below.
Section 1								
8.	Australia 10	1.1	Add principle 9 to the footnote	The text refers to 7 principles, but the footnote has only 6 principles	X			
9.	France 1	1.1	In addition, this Safety Guide ~~supplements and elaborates~~ provides recommendations on meeting the safety requirements on utilization and modification that are established in the A.I.E.A Safety standards Series SSR-3.	As the document is a Safety guide, it should be better to be clear on the fact that recommendations, and not requirements, are provided in the guide.	X			
10.	Pakistan 1	1.1	Please include titles of seven IAEA Safety Fundamental Principles.	To make in line with para 1.1 of DS 510B: Safety in the Utilization and Modification of Research Reactors (Revision of SSG-24).	X			
11.	Germany 1	1.1	Is there a reason to label the principles without numbers 4 and 7?	n/a				Principles 4 and 7 are not specifically addressed in SSG-20. The text is taken as it is no change has been made to original text.
12.	Korea 1	1.1 Footnote2, Page 1	Principle 9 of IAEA Safety Standard Series No. SF-1 shall be added in the footnote of page 1.	Among the seven Principles of SF-1, Principle 9 is not listed in the footnote.	X			
13.	Germany 6 NUSSC	Page 7/ footnote 2	Principle 9 is missing. Add: “Principle 9: Emergency prepared-ness and response: Arrangements must be made for emergency preparedness and response for nuclear or radiation incidents.”	For consistency with the main text and clarity.	X			

14.	Germany 1 RASSC	1.5		In order to explain the difference between the terms “safety analysis” and “safety assessment”, Req. 1 and 5 from SSR-3 should be explicitly cited.	X			
15.	Germany 2 RASSC	1.5		Clarification: The meaning of the following phrase is not clear: “In general, in the safety standards for research reactors the term ‘safety assessment’ is used instead of the term ‘safety analysis’, which has a more specific meaning.”. It suggests that both terms are used to describe the same practice.	X			The text has been deleted.
16.	Germany 3 RASSC	1.7		In the first half of the sentence, the document SSR-3 is cited without giving the specific paragraph. Please add the appropriate paragraphs.	X			
17.	USA 1	1.7,4	Add reference to para 3.1-3.3 of SSR-3.	Reference should be more detailed	X			
18.	Pakistan 2	1.9	Recommendations on safety analyses for experiments at research reactors and experimental facilities are provided in IAEA Specific Safety Guide No. **SSG-510B**, Safety in the Utilization and Modification of Research Reactors Ref. [6] and also modify from the reference list and para 2.44 line 12.	SSG-24 will be superseded by DS 510B so the title/document number may be modified.	X			(in revision as DS-510B) Added in the references.
19.	USA 2	1.9, 1 And 1.10, 2		There appears to be some inconsistency between what these two paragraphs say in relation to experimental facilities. Para 1.10 says they are part of the reactor while para 1.9 directs the reader towards SSG-24. Please clarify to reflect a consistent message				Experimental facilities are covered as a part of research reactors in general in SSG-20 and experiments/ experimental facilities individually are covered in SSG-24 for detailed

								analysis.
20.	Australia 1	1.10	Add "may" to second sentence such that is reads "In this Safety Guide, the term 'research reactor' may include …".	In Australia (and possibly some other Member States), the reactor licence and the licence covering neutron beam instruments are separate and the SAR for the reactor does not cover these instruments.		see also response to France # 2		To be consistent with IAEA safety glossary
21.	France 2	1.10	~~In this Safety Guide, the term 'research reactor' includes associated experimental facilities, critical facilities and subcritical assemblies. An experimental facility includes any device installed in or around a reactor to utilize the neutron flux and ionizing radiation from the reactor for research, development, isotope production or any other purpose.~~ The definition of the term "research reactor" is the one given in SSR-3.	The proposed new text aims at not giving different definitions from one document to another.	X			
22.	France 3	1.11	The amount of detail required in the safety analysis report for small research reactors (i.e. those with a capacity of less than a few tens of kilowatts), and critical ~~facilities~~ and subcritical assemblies may be substantially less.	SSR-3 (§1.3) refers to "critical assembly" and not to critical facility.	X			
23.	Germany 7	1.11	This Safety Guide focuses mainly on research reactors of a capacity of up to a few tens of megawatts. The amount of detail required in the safety analysis report for small research reactors (i.e. those with a capacity of less than a few tens of kilowatts), and critical facilities and subcritical assemblies may be substantially less. Nevertheless, when using the graded approach, all items included in this Safety Guide should be assessed. Hereafter, ~~sub~~critical assemblies will be mentioned	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). It is assumed that critical assemblies are meant here.		As per technical contents provided in SSR-3 guidance is provided, subcritical assemblies are meant here.		The approach of developing guidance that covers all research reactors and sub-critical assemblies is the same as it was followed in development of SSR-3. The approach was also described in the

			separately only if a specific recommendation is not relevant for, or is applicable only to, ~~sub~~critical assemblies. Additional recommendations on the safety analysis, on preparation of the safety analysis report and on the licensing process for high powered or otherwise advanced or complex research reactors are provided in IAEA Safety Guides for power reactors, IAEA Safety Standards Series No. SSG-2, Deterministic Safety Analysis for Nuclear Power Plants [7] and IAEA Safety Standards Series No. GS-G-4.1, Format and Contents of the Safety Analysis Report for Nuclear Power Plants [8]. Use of the Safety Guides for power reactors also necessitates that a graded approach (see SSR-3 [2], paras 2.15–2.17;.SSG-22 [3]) be applied in implementing the recommendations on the basis of the potential hazard associated with the research reactor.					DPP of the Safety Guides. The guidance unless specifically mentioned is applicable also to subcritical assemblies with use of a graded approach that commensurate with their potential risk, as described in the Guides. In addition, there will be also a SSG on use of graded approach.
24.	Vietnam 1	1.11	“This Safety Guide focuses mainly on research reactors of a capacity of up to a few tens of megawatts.” => “less than 30 megawatts" should be specified.	“This Safety Guide focuses mainly on research reactors of a capacity less than 30 megawatts”			X	The current text in SSG-20 is consistent with para 1.8 of SSR-3.
25.	USA 3	1.12, 6	Change “not discussed in” to “not discussed in detail in”	Licensing of decommissioning is discussed. For example, para 2.47	X			
26.	Canada 2	1.13	“Most research reactors have a ~~small~~ significantly lower potential for hazard to the public compared with power reactors, but..”	Dependent on fuel type, enrichment and how the reactor is configured can still have significant potential for offsite consequences. It is agreed that the hazards to the public are lower than for an NPP, but the term “small” is too definitive for a large range of reactor types and uses.	X			

27.	USA 4	1.13, 3	"…operating personnel and experimenters."	Saying operating personnel is too narrow.	X			
28.	Germany 5	1.20	Annexes I and II outline, and provide information on, the application of a basic approach to performing the safety analysis for a research reactor using mainly deterministic methods ~~(which are normally used for safety evaluations for research reactors, however, probabilistic techniques could be used to supplement deterministic methods)~~ to analyze accidents, including their radiological consequences. <u>In addition, probabilistic safety analyses may provide further insights to identify potential improvements for nuclear safety and should be taken into account if reasonably practicable.</u> Annex III deals with specific aspects of the reactor to be described in the safety analysis report. Finally, Annex IV provides a list of typical sources of radiation in a research reactor to be considered and described in the safety analysis report.	During the last years PSA have been applied to several research reactors providing important insights for nuclear safety. Thus, it would be recommended to formulate the application of PSA more demanding rather the explanation in the brackets.	X			
Section 2								
29.	Germany 8	2.6	In some licensing regimes, consideration has been given to the adaptation of a 'pre-licensing' process~~, such as steps that provide for early approval of siting, approval of the safety concept and design, and issuing of a construction licence~~. <u>The pre-licensing process contributes to foster the mutual understanding of licensees, vendors and regulatory body on the design concept, safety concepts as well as safety expectations and requirements to be fulfilled.</u> Such a licensing regime may help	It should be made clear, that "pre-licensing" is not a legal licensing step. The main purpose is to foster a mutual understanding between the future operating organization, the vendor and the regulatory body to ensure, that a certain research reactor project would be licensable. However, "pre-licensing" does not aim to provide a guarantee to issue a construction or operating license, because this would need a thorough review and assessment of the PSAR or	X			See also response to USA comment #5

			to minimize the duplication of effort through different stages of the licensing process. It may also allow for some stages to be conducted in parallel. It provides for the clear division of responsibilities for different stages between regulatory bodies, vendors and operating organizations; gives the public opportunities for early participation; and ensures that the most important safety issues are dealt with early in such a 'pre-licensing' phase. When applying for the construction license, ~~A~~ detailed demonstration of nuclear safety, including an adequate safety analysis, should be submitted in form of a preliminary safety analysis report (PSAR) by the operating organization, and should be reviewed and assessed by the regulatory body before the next stage is authorized. Detailed guidance on the licensing process is presented in SSG-12 [10].	FSAR, which is usually not available in such an early stage of a project. Furthermore "pre-licensing" has not to be confused with issuing the site license.				
30.	USA 5	2.6, 9	Move sentence "A detailed demonstration……" to beginning of para	Introduce basic requirement before discussing pre-licensing	X			
31.	Germany 9	New paragraph	It is common practice to develop different versions of the safety analysis report for different licensing stages of a research reactor as recommended in SSG-12 "Licensing of Nuclear Installations". There are typically three report development stages, as follows: • Initial safety analysis report, which includes the basis for the site authorization; • Preliminary safety analysis report (often abbreviated to PSAR), which includes the basis for the authorization of the construction; • Pre-operational safety analysis report,	It is recommended to add a paragraph similar to para 2.5 from DS449 explaining the different versions of the SARs for the different licensing steps to be in line with SSG-12 which is applicable for research reactors too.				SSG-12 is referred in para 2.6 for detailed guidance on licensing process.

			which includes the basis for the authorization of the commissioning and operation of the nuclear power plant. • During operation of the nuclear power plant, the pre-operational safety analysis report should be further complemented by additional information, leading to the issue of the operational safety analysis report or final safety analysis report (often abbreviated to FSAR). The structure of the safety analysis report proposed in this Safety Guide is best suited to the preliminary, pre-operational and final safety analysis reports. Nevertheless, the same structure of the safety analysis report should be maintained, as far as possible, throughout its development from the initial safety analysis report up to the pre-operational safety analysis report.					
32.	Pakistan 3	2.12	…operated, utilized, modified, extended shutdown and decommissioned without undue radiation risks to site personnel, the public or the environment.	The extended shutdown should also be considered as certain requirements exist for research reactor in extended shutdown as it is also mentioned at requirement 87 of IAEA SSR 3.	X			
33.	Germany 2	2.13, 12	Whether this information is accurate; this might be determined by means of independent ~~verification~~ checks of the design, including calculations, and		X			
34.	France 3	2.15	Design extension conditions ~~are the postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits. Design extension conditions comprise conditions in events without significant fuel degradation and~~	Design extension conditions are defined in SSR-3. It should be better to stay consistent with this definition.	X			

			~~conditions in events with melting of the reactor core~~ defined in SSR-3					
35.	Germany 4	2.16, line 2-4	Examples include maximum allowable doses to the public or the prevention of fuel failure as described in IAEA Safety Analysis for Research Reactors, Safety Reports Series No. 55 [15].	The document cited in this Para should be mentioned as it is the first occurrence of the document (in accordance to the style of the document).			X	In IAEA style of writing references, only safety standards are referred with complete titles. The other documents are referred with reference number only.
36.	Japan 2	2.17	For design extension conditions, the acceptance criteria should provide assurance that the design of the facility is such as to prevent design extension conditions from progressing into early radioactive release or large radioactive release, or to mitigate their consequences, as far as is reasonably practicable in accordance with para ~~6.69~~ 6.24 of SSR-3 [2]. The analysis may lead to implementation of additional safety features, or extension of the capability of safety systems to maintain the main safety functions.	Para 6.69 of SSR-3 does not address design extension conditions.	X			See response to Korea comment #2
37.	Germany 10	2.17	~~For design extension conditions, the acceptance criteria should provide assurance that the design of the facility is such as to prevent design extension conditions from progressing into early radioactive release or large radioactive release, or to mitigate their consequences, as far as is reasonably practicable in accordance with para 6.69 of SSR-3 [2].~~ Acceptance criteria for design extension conditions without significant core degradation should be defined to ensure no	The notion of DEC is appreciated. However, this text is somewhat misleading and should be focused on meeting specific radiological criteria by defining additional technical acceptance criteria.		Acceptance criteria for design extension conditions without significant core degradation should be defined to ensure with adequate level of confidence that core melting can be prevented, that there are adequate		To include avoidance of cliff edge effects; and to avoid repetition of statements on analysis of DEC

			off-site radiological impact or only minor radiological impact. Acceptance criteria for design extension conditions with melting of the core should be defined in such a way that only limited protective off-site measures in area and time will be necessary. Early releases and large releases have to be practically eliminated anyhow. The analysis may lead to implementation of additional safety features, or extension of the capability of safety systems to maintain the main safety functions.			margins to avoid cliff edge effects, and there is no, or only minor, off-site radiological impact Acceptance criteria for design extension conditions with core melting should be defined in a way that ensures mitigation of consequences as far as reasonably practicable, and early and large radioactive releases are practically eliminated in accordance with para 6.68 of SSR-3[2].		
38.	USA 6	2.17, 3	Add footnote to explain early release	SSR-3 has such a footnote	X			
39.	Korea 2	2.17, Pages 12-13	2.17. For design extension conditions, the acceptance criteria should provide assurance that the design of the facility is such as ~ to mitigate their consequences, as far as is reasonably practicable in accordance with para ~~6.69~~ 6.68 of SSR-3 [2].	Since the para. 6.69 of SSR-3 is about the combinations of events and failures, it should be corrected to para 6.68.	X			
40.	Germany 3	2.18a	Dose limits (or design target doses) for ~~staff of the operating organization~~ site personnel, ~~including experimenters and workers at the reactor site;~~	The IAEA glossary does not list 'personnel' only, it is site, external or operating personnel. Site personnel includes all persons.	X			

				What is meant here? All personnel?				
41.	Canada 3	2.18	In the development of the specific acceptance criteria, consideration should be given to the criteria listed below as appropriate for the type of the facility:	New text retains key message from text removed from 2.19 in next comment.	X			
42.	Australia 12	2.18, (b)	Remove last dot point	It's unclear what "Maximum damage of fuel assemblies in the core" means in the context of the paragraph	X			
43.	USA 7	2.18 (b)	Add "Maximum fuel and cladding temperature below failure"	To account for TRIGA design where failure is based on fuel phase change or gas pressure in fuel-clad gap that exceeds ultimate strength of cladding.	X			
44.	Korea 3	2.18 (b)	~~Frequency limits for significant damage to~~ Prevention from systematic failure of fuel cladding	Frequency limits are not appropriate and cannot be developed for nuclear fuel performance criteria, instead, fuel cladding should be prevented from systematic failure (due to fretting, hydride, oxidation, etc.)		...Limits for significant damage and number of fuel cladding failure;		
45.	Korea 4	2.18 (b) Page 13	~~Maximum~~ Limitation of significant damage of fuel elements in the core resulting in an early or a large radioactive release.	'Maximum damage' is not appropriate for nuclear fuel performance criterion. It seems to be added considering DEC, unless, needs to be corrected by another appropriate criterion		Deleted		See response to Australia Comment 12
46.	Germany 11	2.19	~~Some of the acceptance criteria mentioned above may not be applicable to low power research reactors, critical facilities and subcritical assemblies, depending on their specific designs. Additionally, for subcritical assemblies, there may be acceptance criteria specified for limits on insertion of reactivity that prevent criticality.~~	Paragraph 2.18 provides a guidance how acceptance criteria may look like. Thus, para. 2.19 is not necessary and doesn't provide any further guidance. Always, specific criteria have to be developed for a specific design.			X	It provides guidance especially for subcritical assemblies, see response to Canada comment #4 the para has been modified.
47.	Canada 4	2.19	Where ~~Some of the~~ specific acceptance criteria mentioned above are determined may not to be applicable to low power	The clause as written does not indicate that the proponent should at least go through the exercise of confirming which	X			

			research reactors, critical facilities and subcritical assemblies, ~~depending on their specific designs~~ the rationale should be documented. Additionally, for subcritical assemblies, there may be acceptance criteria specified for limits on insertion of reactivity that prevent criticality.	criteria in 2.18 are applicable or not or why in the safety analysis report. It is not enough to present final decisions….how one arrives at those decisions should be traceable.					
48.	Japan 3	2.20, last sentence	Systems used for mitigation of the consequences of accidents should be designed and constructed, depending on the importance to safety, to withstand the maximum loads and stresses and the most extreme environmental conditions for the accident analysed.	Applying “most extreme environmental conditions” on the mitigation systems is too strict. They should be designed in accoudance with the importance to safety.	X				
49.	Japan 4	2.24	For research reactors with low potential hazard, particularly critical facilities and subcritical assemblies, the amount of information and analysis to be provided according to paras. 2.26 and 2.48 ~~can be substantially~~ could be reduced in accordance with a graded approach.	Intentional judgment should be avoided.	X				
50.	Germany 12	2.24	For research reactors with low potential hazard, particularly critical facilities ~~and subcritical assemblies~~, the amount of information and analysis to be provided according to paras. 2.26 and 2.48 can be substantially reduced.	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). For that reason, Sub-critical assemblies shall be deleted here.				X	The approach of developing guidance that covers all research reactors and sub-critical assemblies is the same as it was followed in development of SSR-3. The approach was also described in the DPP of the Safety Guides. The guidance unless specifically mentioned is

								applicable also to subcritical assemblies with use of a graded approach that commensurate with their potential risk, as described in the Guides. In addition, there will be also a SSG on use of graded approach.
51.	Canada 5	2.24	Delete entire clause: ~~For research reactors with low potential hazard, particularly critical facilities and subcritical assemblies, the amount of information and analysis to be provided according to paras. 2.26 and 2.48 can be substantially reduced.~~	It is agreed that the graded approach can and should be applied, but this should be reflected in paragraphs 2.26 and 2.48. (see comment 5 and 6) The analysis of risk should determine the level of detail needed and the proponent should justify the level of detail is appropriate. This is particularly important when use of elements of the SAR in public licensing discussions becomes more pervasive. This clause, as written, implies that the proponent has the ability to pre-judge before performing the necessary analyses ("potential hazard")			X	Updated the text Other Member States has provided comments to improve the text. Please see responses to Japan comment #4 and Germany comment #12
52.	Canada 6	2.26	The operating organization should provide sufficient information commensurate with the novelty, complexity and potential for harm posed by the facility to demonstrate to the regulatory body that the proposed site is suitable for the type and design of the proposed research reactor	See comment 4. Addresses the removal of 2.24.	X			

53.	Finland 1	2.28	Consideration should also be given to nuclear security, including physical protection system and information security, and interface with safety.	For completeness.	X			
54.	Korea 5	2.29, line 10	This information should be included in the safety analysis **report** subject to updating as design and construction proceed.	The term 'report' needs to be added for clarity.	X			
55.	Germany 13	2.33, line 1	Commissioning tests shall be arranged in functional groups and in a logical sequence. This sequence includes pre-operational tests, initial criticality tests, low power tests, and power ascension and power tests.	Wrong citation	X			
56.	Japan 5	2.33./ Line 6-9	The test results should be approved by the operating organizaition at the appropriate level of management and, as neccesary, by the regulatory body ~~as necessary~~ before the subsequent test sequence is started.	Clarification. It can be interpreted that the approval of the operating organization is also "as necessary".	X			
57.	Germany 14	2.34	For subcritical assemblies, ~~initial criticality tests and~~ low power tests of Stage B and tests of Stage C (power ascension and full power) are not applicable. For subcritical assemblies ~~However,~~ tests should be performed to verify that the configuration is subcritical without the external neutron source. Some other tests, such as approach to criticality and neutron flux measurements are also needed. Such tests and measurements should be used to validate the computational models and tools that are used for design and safety analysis of the subcritical assemblies.	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). It is assumed that critical assemblies are meant here.			X	Consistency with Stage B and Stage C of commissioning. "…. To verify that the configuration is subcritical" is more technically accurate and precise, and it is adequate.
58.	USA 8	2.36, 2	Replace "the next stage" with "that next stage"	To clarify that the review is performed before the next stage starts.	X			
59.	Germany 5	2.37, line 1	Stage A ~~(test prior to fuel loading)~~	The part can be deleted in accordance to para 2.38.	X			

60.	Germany 15	2.38, line 1, page 18	Stage B (Fuel loading tests, initial criticality tests and low power tests ~~loading of fuel and initial criticality~~)	For consistency with the commissioning stages A to C described in NS-G-4.1.		Throughout the text, the detail description of commissioning Stages has been removed, as it is described in the earlier paragraph (see also response to USA comment #9)		To have consistent parallel construction of the text.
61.	USA 9	2.38, 1	Reverse deletion of short description of stage B or remove short description of stage A	To have consistent parallel construction of the text	X			See also response to Germany comment #15
62.	Korea 6	2.39, Page 19	2.39. As power ascension test and ~~full~~ power test processes in Stage C move closer to completion, ~ and other occurrences.	The name of test in Stage C shall be corrected for consistency.		Please see responses Germany #15 and USA # 9.		To have consistent parallel construction of the text.
63.	Korea 7	2.42, Page 20	The need for review may arise in a number of ways, such as periodic safety reviews required by the regulatory body or self-assessments performed by the operating organization. (The requirements are given in 7.121 and 7.122 of SSR-3)	It is preferred to refer the relevant requirement of SSR-3 with respect to PSR.	X			
64.	Canada 7	2.48	At some point in the decommissioning process (e.g. after the removal of all fuel from the site), the risk profile of the facility becomes low enough that the safety analysis report ceases to be a major working document and a sufficiently detailed report on the decommissioning process should be prepared commensurate with the remaining hazards. Further guidance on decommissioning is provided in IAEA Safety Standards Series No. Ref.SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [923].	See comment 4. Addresses the removal of 2.24.				Although, the proposed revisions are technically correct, they are implicitly included in the text-the focus of the whole chapter is the information to be submitted for review and assessment.

Section 3								
65.	Japan 1 EPReSC	3.2	To aid in the development of ~~the emergency preparedness and response arrangements~~ emergency arrangements.	A proper wording based on SSR-3 and GSR Part 7.	X			
66.	USA 10	3.2, 9 (third dash)	Replace “appreciation” with “understanding”	Stronger word	X			
67.	USA 11	3.2, general	Consider adding dash “to aid in the understanding of the interaction between safety and security”	Add important aspect of facility design		to aid in the understanding of the ~~interaction~~ interface between safety and security		
68.	USA 12	3.3, second dash	Consider expanding this dash to more than decommissioning. “Events that may have occurred during the lifetime of the research reactor (or operating experience feedback, including from other nuclear installations). These may give rise to changes in the facility and its operation and may influence the actions that will need to be taken during the eventual decommissioning of the research reactor.”	Added parenthetical shows greater need for updates that just decommissioning	X			
69.	Australia 2	3.4	Similar to above, wording should reflect the possibility that neutron beam instruments may be covered under a separate licence to that covering the reactor.	See above comment on 1.10			X	Para 3.4 refers to safety analysis not to the licensing process,
70.	Korea 8	3.4, Page 24, 3.11 Page 25	3.4. Detailed description that shall be included in the safety analysis report are discussed in the para. 3.6 of SSR-3 [2]. ~~The safety analysis report should give a detailed description of the research reactor site, the research reactor itself, the experimental facilities and devices and all other facilities with significance for safety. It should provide a detailed description of the general~~	Most statements of the para. 3.4 and 3.11 are identical to the para. 3.6 and 3.9 of SSR-3, respectively, except these are expressed as “should” statement instead of “shall” as in SSR-3. According to the DDP section 3.2, these statements should be either removed from the guide or modified to make them useful recommendations.		“Reference to para 3.6, Requirement 1 of SSR-3 [2]” is provide for paragraph 3.4 For paragraph 3.11, requirement 1 para 3.9 has been quoted		The wording mentioned in para 3.4 of this Safety Guide is not exactly the same as of the requirement 1 para 3.6, so the reference is

			~~safety concepts and criteria, as well as of the codes and standards applied to the design for the purposes of protection of the reactor, the operating personnel, the research reactor users, the public and the environment. The potential hazards associated with the operation of the research reactor should also be addressed in the safety analysis report.~~ 3.11. Requirements on the references of the safety analysis report are discussed in the para. 3.9 of SSR-3. ~~The safety analysis report should present adequate references that may be necessary for the review and assessment process. This reference material should be freely available to the regulatory body and should not be subject to any classification or limitation that would prevent its adequate review and assessment~~			with shall statement.		provided. For para 3.11, the wording was the same so the requirement is quoted with shall statement.
71.	Japan 2 EPReSC	3.5	The safety analysis report should also provide details of the emergency ~~preparedness and response~~ plan, and decommissioning plan.	A proper wording based on SSR-3 and GSR Part 7.	X			
72.	USA 13	3.5, 2	Change “for operation, or” to “for operation and their bases, or”	Justifying the operational limits and conditions is an important function of the SAR	X			
73.	Japan 3, EPReSC	3.6	However, some of the topics may be discussed in separate documents (e.g. in the operational limits and conditions, operational ~~and emergency preparedness and response arrangements~~ procedures, physical protection plans, emergency plans and procedures and decommissioning plan).	A proper wording based on SSR-3 and GSR Part 7.	X			
74.	Germany 16	3.7	The operating organization should ensure that an independent verification of the safety assessment is performed by individuals or	The main purpose for submitting the SAR is for review and assessment by the regulatory body. Not the SAR will be	X			

			groups separate from those carrying out the design, before the safety analysis report is submitted to the regulatory body for review and assessment as part of a licensing procedure ~~licensing~~ (see Requirement 5 of Ref.SSR-3 [2]).	licensed, but the research reactor.				
75.	Germany 17	3.8	The independent verification should be carried out under the responsibility of the operating organization by a team of experts who should be independent of the designers and of those performing the safety assessment. Personnel are considered independent if they have not participated in any part of the design or the safety assessment. This independent verification is in addition to the reviews carried out within the design organization. Usually the SAR is prepared by the vendor. Thus, reviewing the SAR by the operating organization contributes to the familiarization of the operating organization with the design of the research reactor.	As usually the SAR is prepared by the vendor, the independent review by the licensee also contributes to the familiarization with the design.			X	The text in para is about the need for independent verification. The benefit to the operating organization is a separate topic.
76.	Australia 3	3.10	Note for information only: based on Australian experience, producing a public version of the SAR involves significant effort and resources. There may also be a need to provide legal justification for the removal of every individual piece of information.		n/a			
77.	Germany 18	3.10	~~In some States~~ The proposal and the licence application for a research reactor should ~~may~~ be subject to public participation by means of regular meetings, formal hearings or other appropriate means of communication ~~an open public debate~~. For these purposes, the operating organization may have to develop a non-technical version of the safety analysis report that can be understood by the public,	Public debate seems not to be an appropriate term. Moreover, public participation is recommended in SSG-12. With the proposed modification para. 3.10 is more in line with SSG-12 (e.g. para. 2.44).	X			

			considering confidentiality aspects. Guidance on public participation is given in SSG-12 [10].					
78.	Germany 19	3.16	For small, low risk facilities (such as ~~subcritical assemblies,~~ critical facilities, or research reactors with low power levels), these requirements are much less stringent. However, as the safety analysis report is often the only comprehensive document produced, every topic discussed in the Appendix to this Safety Guide should be considered. Although the extent of information on each topic would be limited, the scope of some topics (e.g. the protection of operating personnel against overexposure in critical assembly facilities) may be much larger for small, low power facilities.	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). Subcritical assemblies shall be deleted here.		For low risk facilities (such as some critical assemblies, subcritical assemblies, or research reactors ……		The approach of developing guidance that covers all research reactors and sub-critical assemblies is the same as it was followed in development of SSR-3. The approach was also described in the DPP of the Safety Guides. The guidance unless specifically mentioned is applicable also to subcritical assemblies with use of a graded approach that commensurate with their potential risk, as described in the Guides. In addition, there will be also a SSG on use of graded approach.
79.	Korea 9	3.18, Page 29	The safety analysis should also serve as a basis for the determination of the operational limits and conditions, the safety	It needs to be corrected for clarity (…a basis for A, B, C and D)	X			

			classification ~~for~~ of the structures, systems and components, ~~for~~ and the development of the accident management procedure and ~~for~~ the emergency preparedness and response plan.					
80.	USA 14	3.20, First dash	Change to "That sufficient defense in depth has been provided, and that the levels of defense are preserved to the extent possible in that potential accident sequences are arrested as early as possible."	The rewording has left the idea of preserving defense in depth without explanation.	X			
81.	Japan 6	3.20./ 2nd Bullet	That the research reactor can withstand the physical and environmental conditions that would experience. These ~~is~~ would include ~~robustness against~~ extreme environmental conditions and other extreme conditions.	It would be too strict to require robustness against extreme environmental conditions and other extreme conditions.	X			
82.	USA 15	3.20, 4th dash, line 4	Change "do not increase." to "do not unacceptably increase."	There should be allowance that the design may allow an increase in risk as long as the increase is acceptable	X			
83.	UK 1	3.21	Extra text to be added to clarify if events are for design basis, AOO etc or for any events which can challenge the full extent of defence in depth	The need to identify postulated events through e.g. HAZOPs and FMEA is listed. However, it is not stated whether these are for DBA, AOO etc. This perhaps did not matter previously in earlier version of SSG-20 but there is now a dedicated section on DEC.				No specific text is proposed. Paragraph 3.21 is applicable to identification and selection of PIEs in all facility states. Further guidance is provided in subsequent paragraph.
84.	Australia 4	3.22	Suggest consideration be given to including this list of PIEs in an Annex rather than the body of the Guide	Facilitates users understanding			X	PIEs are sufficiently important to the document to justify retaining them in main text.

85.	USA 16	3.22, (2), 1st dash	Remove parenthetical or make it an example (e.g.)	There could be other events other than fuel insertion (e.g., dropping fuel assembly on core)	X			
86.	USA 17	3.22, (2), 10th dash	Add "error in loading or unloading" to example list	Physical movement of experiments can significantly change reactivity	X			
87.	USA 18	3.22, (5) general	Add "failure of engineered safety features"	There are more ESFs than just the ECCS	X			
88.	USA 19	3.22 (7)	Reorganize list to group natural phenomena	Improves organization to aid the user	X			
89.	USA 20	3.22 (7)	Remove lightning from hurricanes if standalone	Lightning appears twice	X			
90.	USA 21	3.22 (7)	Add snow and ice storms	Results in greater completeness of list	X			
91.	USA 22	3.22 (7)	Add failure of pipelines	Results in greater completeness of list	X			
92.	Japan 7	3.22.	The following list of selected postulated initiating events is based on the appendix to SSR-3 [2]: .... (7) External events - Earthquakes (including seismically induced faulting and landslides); - Flooding (including failure of an upstream or downstream dam and blockage of a river and damage due to a tsunami or high waves); - .... - Volcanic eruption;	The effects on the safety functions due to the volcanic eruption should be considered.	X			
93.	Germany 20	3.22	Add PIE for DEC, e.g. • Anticipated transient without scram	The list of PIEs seems to be restricted to AOO and DBA. Obviously, no DEC are			X	Guidance on analysis of DEC is

			○ Maximum reactivity insertion by withdrawal of control elements on the basis of the operating conditions "full load" ○ Loss of main heat sink with unavailable station service power supply • Loss of energy supply ○ Loss of off-site power cumulated with the failure of all emergency diesel generators • Loss of component cooling ○ Loss of the component cooling water system • Loss of secondary-side heat removal Total loss of secondary site cooling water	included in this list. As required in SSR-3 Requirement 18 PIEs should cover all accident conditions which include DECs.				already provided, and it is not appropriate to develop a specific list on PIEs for DEC.
94.	UK 2	3.22	"Typical examples of postulated initiating events leading to event sequences categorized as design basis accidents should include those given below, sorted by types of sequence. This list is broadly indicative. The actual list will depend on the type of reactor and actual design:"	Despite 3.21 stating the HAZOPs and FMEA should be used, it is not stated in 3.22 if the list from SSR-3 is definitive and/or prescriptive. Wording from the recent update to SSG-2 could be used (para 3.30)	X			
95.	Canada 8	3.23	The list of PIEs specified in para 3.22 should be reviewed for applicability for subcritical assemblies ~~and may be reduced significantly~~. The resultant list of PIEs should be justified and documented for the specific facility configuration. For example, the following PIEs may not be applicable to some subcritical assemblies, depending on their specific design features:…	Deleted text is not necessary to articulate the guidance. One item that is missing (see red text) is that a good practice is to ensure documented traceability of the justification of the PIE list. This additional text introduces guidance in subsequent clauses.	X			
96.	Germany 21	3.24	Postulated initiating events should be grouped ~~categorized~~ in accordance with their expected frequency of the initiating events and clearly	Para. 3.24. is not in line with the general approach of categorizing PIEs. PIEs are usually assigned to the different plant states AOO, DBA and DEC mostly		Accepted but the remaining text is not deleted		To ensure coherence with guidance on nuclear power

			assigned to the different plant states. ~~anticipated system response. The purpose of this grouping categorization is:~~ ~~— To justify the basis for the range of events under consideration;~~ ~~— To reduce the~~ The number of initiating events requiring detailed safety analysis can be reduced by ~~to a set that includes the~~ enveloping cases in each of the various event groups. ~~credited in the safety analysis but that does not contain events that are associated with identical system performance (such as events that are identical in terms of timing, plant systems response and radiological release fractions);~~ — ~~To allow for d~~ Different acceptance criteria for the safety analysis should ~~to~~ be applied to different plant states ~~event classes~~.	based on their frequency. In addition, para. 3.24 is not in line with SSG-2 chapter 2. There is no technical reason to choose a different methodology to categorize PIEs.				plants. The remaining text is kept for guidance.
97.	Germany 22	3.25	Both internal and external postulated initiating events of all types, for all operational states, including shutdown and fuel loading, should be considered in this process of event grouping ~~classification~~. The process of event grouping ~~classification~~ should lead to a list of enveloping postulated initiating events to be analysed. Failures in other systems such as experimental facilities, failures in the availability of off-site power or the total loss of off-site power, and failures in spent fuel storage and in storage tanks for radioactive liquids should also be considered.	See comment on para 3.24. The term "grouping" seems to be more adequate than "classification" also considering terminology of SSG-2.	X			
98.	Germany 23	3.26	In the selection~~, categorization~~ and grouping ~~classification~~ of postulated initiating events for the analysis, the list given in para. 3.22 should form the basis of the postulated	See comment on para 3.24. The term "grouping" seems to be more adequate than "classification" also considering terminology of SSG-2.	X			

			initiating events to be considered. (…)					
99.	Finland 2	3.28	The safety analysis should identify design basis accidents and design extension conditions in events without significant fuel degradation and with melting of the reactor core. In addition, accidents beyond the design ~~basis~~ envelope that have more severe consequences may be analysed for purposes of emergency planning and for specifying the measures to be taken to mitigate the consequences of an accident	Please clarify: accidents beyond the design ~~basis~~ envelope, accident more severe than considered in the design in line with GSR Part 4. As well design extension conditions with melting of the reactor core should be included to demonstrate that early and large releases have been practically eliminated as required in 3.35.		"Beyond design basis" deleted		Consistency with glossary and EPR requirements.
100.	Germany 24	3.29	Annex I deals mainly with deterministic methods, which are normally used for safety assessments of research reactors. Deterministic techniques for anticipated operational occurrences and design basis accidents ~~research reactors~~ are characterized by conservatism and are based on defined sets of rules for event selection, analytical methods, and parameter specification and acceptance criteria. For design extension conditions best estimate methods with realistic boundary conditions can be applied. Through the use of these methods, reasonable assurance is provided that the ultimate objective of preventing or limiting the release of radioactive material can be achieved without the need to perform complex calculations, because these methods tend to overestimate the amount of radioactive releases. The most severe of these releases are taken into account in the selection of a site or in setting design requirements for engineered safety features for the research reactor. The choice of these accidents is based on experience and	Para 3.29 does not differentiate between different deterministic methods for the different plant states. Whereas conservative methods are appropriate for DBA, best estimate methods are more appropriate for DEC (see discussion in DS491).	X			

			engineering judgement, without the benefit of determining the probabilities of the event sequences.					
101.	Germany 6	3.31 (a)	Postulated initiating events that are likely to occur during the lifetime of a research reactor, but do not lead to accident conditions (anticipated operational occurrences), ~~which~~ should be analysed to show that the research reactor has a sufficient safety margin to comply with the acceptance criteria for such events.		X			
102.	Germany 25	3.33, Page 36	~~DEVELOPMENT OF SAFETY ANALYSIS FOR DESIGN EXTENSION CONDITIONS~~	Headline before para. 3.33. can be deleted, as a similar headline for DBA and AOO doesn't exist.	X			
103.	Australia 13	3.33	After "that could lead to reactor core damage" consider adding "or some other radiological release" (or similar)	The para implies DECs are only related to core damage. At some research reactors, DECs could also be related to e.g. target damage, major tritiated heavy water releases etc., i.e. events other than core damage.	X			
104.	Germany 7 RASSC	3.33 first line	Requirement 22 of SSR-3 [2]	Missing reference	X			
105.	France 4	3.35	Analysis of design extension conditions should also demonstrate that • The reactor can be brought into the state where the confinement function can be maintained in the long term; • The structures, systems and components are capable of avoiding an early radioactive release or a large radioactive release; ~~• The possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated~~ and; • Control locations remain habitable to allow performance of required actions.	According to SSR-3 and French practices, practically eliminated situation are not part of DEC (see 6.68 of SSR-3 that makes a difference between: • the practical elimination of the possibility of certain conditions • and the objective of DEC conditions)	X			

			In addition, it should be demonstrated that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated. Besides, additional accidents that are postulated for the purposes of emergency preparedness and response should be analyzed.					
106.	USA 23	3.37	This appears not to be consistent with SSR-3 para 6.66 which refers to designs where criticality would not be a design extension event but part of the design basis. Treating criticality as a design basis event should be considered and the document revised accordingly.	The thought processes of the authors of this new section and how they related it to SSR-3, para 6.66 are not known		"As stated in para 6.66 of SSR-3 [2], "for subcritical assemblies, likelihood of criticality shall be sufficiently remote to be considered as a design extension condition". This event should be analyzed to demonstrate compliance with pre-established acceptance criteria and to		The para is now consistent with SSR-3

						ensure adequate margins to avoid any cliff edge effects as well as to identify additional safety features to prevent or mitigate the consequences of such event”		
Section 4								
107.	Germany 26	4.3/ line 5/Page 39	paras 2.~~26~~ 25–2.48.	Para. 2.25. “Schedule for the submission of information” should be part of the planning of the review and assessment programme.	X			
108.	Finland 4	4.6	. The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for the construction of the research reactor should include: (a) The competence and capability of the operating organization to meet the licence requirements; (b) The site characteristics, to confirm the acceptability of the site and the related data used in the design of the proposed research reactor; (c) The basic design of the proposed research reactor, to confirm that it will meet the safety requirements, including	Physical protection is not enough. also, information security, including computer security should be considered.	X			

			requirements for occupational health and 40 requirements for fire safety; (d) The management systems of the operating organization and those of its vendors; (e) The design features ~~relating to~~ of the nuclear security system (including physical and information security) that are important to safety; (f) Information necessary for verification of the design.					
109.	Finland 5	4.8	The documents that should be submitted to the regulatory body for review and assessment in order to obtain authorization for commissioning Stage B (loading of fuel and initial criticality) should include: (a) The records of the results of the previous commissioning stage, including non-conformances and, where appropriate, their associated corrective actions; (b) The revisions to the commissioning programme, if any; (c) The operational limits and conditions for Stage B commissioning; (d) The provisions for radiological protection; (e) The adequacy of the operating instructions, operating procedures, emergency procedures and administrative rules; (f) The records and reporting systems; (g) The training and qualification of research reactor personnel, including the levels of staff and their suitability for the work; (h) The occupational health and fire safety	Please replace physical protection with security. see also 4.6	X			

			aspects; (i) The management system, organization and programme for operation; (j) The emergency plan; (k) The system of accounting for and control of nuclear material and radioactive material; (l) The arrangements for security ~~pPhysical protection~~ plan of the research reactor.					
Appendix								
110.	Germany 27	Page 42, Appendix	The information required for the content of the safety analysis report for ~~sub~~critical assemblies should be the same as for research reactors. However, the amount of information and the level of detail can be substantially reduced with consideration to the lesser complexity and lower hazards of ~~sub~~critical assemblies. In addition, some technical contents of those mentioned in this Appendix may not be applicable to ~~sub~~critical assemblies. Contents that are not applicable to ~~sub~~critical assemblies are highlighted throughout the Appendix by an asterisk (*), or specifically indicated.	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). It is assumed that critical assemblies are meant here.		"some types of subcritical assemblies……" replaced "….subcritical assemblies"		Added in the sentence to clarify about subcritical assemblies. Please also see response to Germany comment # 12, 19
111.	Canada 9	Appendix	The information required for the content of the safety analysis report for subcritical assemblies should be the same as for research reactors. However, consistent with a graded approach, the amount of information and the level of detail ~~can be substantially reduced~~ should be consistent with ~~consideration to the lesser~~ the	It is agreed that the graded approach can and should be applied, but analysis of risk should determine the level of detail needed and the proponent should justify the level of detail is appropriate. This clause, as written, implies that the proponent has the ability to pre-judge before performing basic analyses	X			

			complexity and ~~lower~~ hazards of the specific facility ~~subcritical assemblies~~. In addition, some technical contents of those mentioned in this Appendix may not be applicable to subcritical assemblies. Contents that are not applicable to subcritical assemblies are highlighted throughout the Appendix by an asterisk (*), or specifically indicated.	("potential hazard"). A subcritical nuclear assembly with a subcriticility margin very close to 1 has a significantly different risk profile from one with a margin of 0.5. And the size and use of the facility influences the risk profile as well. The Myrrha prototype project at up to 100 MWth is an example of a fast spectrum concept that is complex enough to potentially warrant a higher level of technical information in some areas.				
112.	UK 3	Appendix	"The section headings of the Appendix are, in general, the headings that may be appropriate for the different chapters of the safety analysis report. Variations, additions and deletions may be necessary depending on the type of reactor and the approach taken to demonstrating safety"	The Appendix on the content of a Safety Analysis Report does say "the headings that may be appropriate for the different chapters of the safety analysis report" suggesting that they may not all be compulsory. However, given that many new research reactors may be exploring new ways of delivering safety (ie Gen IV type designs), more freedom should be given to authors if the technology allows.		…The amount of information and the level of detail may vary depending upon the type of facility…		
Chapter 1								
113.	Pakistan 4	A.1.5	The term "nuclear research facilities" should be replaced with **"research reactor".**	For harmonization of document.		…nuclear facilities…		
Chapter 2								
114.	Germany 29	Page 50 / A.2.3. / line 15	The extent to which redundancy, diversity, physical separation and functional independence	Clarification that "diversity" and "physical separation" are two separated objectives	X			

115.	Germany 31	Page 52 / A.2.4. (9) / line 7	(b) Requirements for coolant system integrity and protection of the boundary from leakage*; (c) Preventing the uncovering of the core*	Critical assemblies might be cooled by air. Therefore, these requirements may not be applicable.	X			
116.	Korea 10	Page 48 Para A.2.4. (18)	(f) Independence and performance of data communications (g) Suitability of pre-developed softwares used in system	Computer based systems and software design should be considered to verify data communications and pre-developed items according to the paragraph 8.20, 8.21, 8.22, 8.25 and 8.26 of IAEA SSG-37	X			
117.	USA 24	A2.4 (5)	Remove *	It is common for sub-critical assemblies to be at research reactors and take advantage of shared facilities. Do not understand why this para has an *	X			
118.	USA 25	A2.5 old (16) and 17(b)	Revise as needed to recognize importance of surveillance	Don't understand why (16) was removed from list. Don't understand why ECCS is only example of surveillance	X			
119.	Canada 10	A. 2.5	~~If any scheme has been devised for the classification of structures, systems and components for purposes of analysis or design, such as for seismic safety or nuclear safety,~~ The basis for ~~the~~ safety classification~~s~~ and the list of classes should be presented in this section of the safety analysis report.	A safety classification methodology (even a simple one) would be applicable regardless of the size and complexity of the facility. It is part of the justification supporting the safety case. It is not clear why the IAEA would suggest otherwise. SSG-30	X			

			Please consider including reference to SSG-30.	would be useful as a reference guide given the the approaches contained in the document would be very similar for research reactors.				
120.	Germany 28	A2.6	(…) Extreme weather conditions including effects due to climate change should be taken into account for the determination of the external events as well as combinations of external events. Additional information on siting requirements is presented in section 5 of Ref.SSR-3 [2].	Extreme weather conditions are not only caused by climate change. These cases should also be covered here.	X			
Chapter 3								
121.	Canada 11	Chapter 3 All.	Is the site evaluation section in SSR-3 being replaced by SSR-1 *Site Evaluation for Nuclear Installations* just recently approved for publication?	Please check the reference is correct.				SSR-3 were published in consistent with other safety requirements.
122.	Japan 8	A.3.2	Information should be provided in sufficient detail to support the analysis and conclusions of Chapter 16 of the safety analysis report, to demonstrate that the research reactor can be safely operated at the proposed site. For many low power research reactors, ~~including~~ critical facilities, and subcritical assemblies, which present very limited hazards,	Clarification To keep a consistency with other paragraphs.	X			

			the amount of detail provided in this chapter can be substantially reduced.					
123.	Germany 30	A.3.2	Information should be provided in sufficient detail to support the analysis and conclusions of Chapter 16 of the safety analysis report, to demonstrate that the research reactor can be safely operated at the proposed site. For many low power research reactors including critical facilities, ~~and subcritical assemblies,~~ which present very limited hazards, the amount of detail provided in this chapter can be substantially reduced. ~~In addition, most of the details described below related to geology and seismology, meteorology, hydrology and oceanography, radiological impact, adequacy of the site for emergency measures may not be required for subcritical assemblies.~~	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). Subcritical assemblies shall be deleted in this paragraph.		"Some of the subcritical assemblies"		Added at the end of the sentence to clarify about subcritical assemblies. Please also see response to Germany comment # 12, 19
124.	Germany 32	A.3.2	Information should be provided in sufficient detail to support the analysis and conclusions of Chapter 16 of the safety analysis report, to demonstrate that the research reactor can be safely operated at the proposed site. For many low power research reactors including critical facilities, ~~and subcritical assemblies,~~	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). Subcritical assemblies shall be deleted in this paragraph.		"…..Some of the subcritical assemblies"		Added at the end of the sentence to clarify about subcritical assemblies. Please also see response to Germany comment # 12, 19

			which present very limited hazards, the amount of detail provided in this chapter can be substantially reduced. ~~In addition, most of the details described below related to geology and seismology, meteorology, hydrology and oceanography, radiological impact, adequacy of the site for emergency measures may not be required for subcritical assemblies.~~					
125.	USA 26	A.3.4 (d)	Add pipelines	Pipelines carrying natural gas and petroleum products can represent a hazard to the facility	X			
126.	USA 27	A.3.4	Add area under regulatory control or licensed area	The licensed area should be described and may differ from the boundaries already discussed in this section	X			
127.	Vietnam 2	A.3.8	Suggest adding a new paragrahap before paragraph A 3.8: "This section should provide information concerning the seismic and tectonic characteristics of the site and of the region surrounding the site. The evaluation of seismic		X			Addressed in same para A.3.8.

			hazards should be based on a suitable geotectonic model substantiated by appropriate evidence and data. The results of this analysis, to be used further in other sections of the SAR in which structural design, seismic qualification of components and safety analysis are considered, should be described in detail".					
128.	Vietnam 3	A.3.17	Suggest adding a new paragraph before the paragraph A 3.17: "This section should cover all aspects of site activity that have the potential to affect the radiological impacts of the site throughout the lifetime of the the reactor, including construction, operation under normal conditions and decommissioning."		X			Covered in same para A. 3.17

129.	Germany 33	A.3.17.	A.3.17. This section should describe radiological aspects and, in particular, the biological aspects of transfers of radioactive material to people. Most of these details may not be required for low hazard, low power reactors~~,~~ and critical facilities ~~and subcritical assemblies~~. In this case, only a brief summary should be given under each heading. If no radiological impact section is provided, justification should be provided for omitting this section of the safety analysis report.	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). Subcritical assemblies shall be deleted in this paragraph.		"Most of these details may not be required for some low hazard, low power reactors, critical assemblies and some of subcritical assemblies"		See also response to Germany comment #30
130.	Vietnam 4	A.3.19	Suggest elaborating the paragraph: "A.3.19. The population distribution around the research reactor and in the region, including seasonal and daily variations, should be presented in this section. In particular, information on existing or projected population distributions around the research reactor should be	"A.3.19. The population distribution ***and land use that is relevant to the safe design and operation of the research reactor and*** around the research reactor and in the region, including seasonal and daily variations, should be presented in this section. In particular, information on existing or projected	X			

			collected and kept up to date during the lifetime of the research reactor.”, to be the new one:	population distributions around the research reactor should be collected and kept up to date during the lifetime of the research reactor.”				
131.	USA 28	A3.22	Consider adding that this section should contain information on where and how the operating organization will obtain information on atmospheric conditions in real time.	This information could be important if a release of radioactive material occurs			X	The section doesn't cover the emergency preparedness for that the operating organization needs information on atmospheric conditions in real time.
Chapter 4								
132.	Germany 8 RASSC	A.4.2	“(air locks, doors, windows, etc.)”	Other building penetrations than the two mentioned in the text are possible and should be included.	X			
133.	Korea 11	Page 57 Para A.4.2. (18)	A.4.2. The description should include the design basis of ~ the building penetrations (air locks, doors, mechanical and electrical penetrations, etc.) in relation to their resistance to internal and external events (see paras A.2.11 and A.3.7).	It is necessary to describe additional penetrations (mechanical and electrical penetrations), which should be also designed for internal/external events	X			
Chapter 5								

134.	USA 29	A5.1, 5	Change “fuel storage” to “fuel storage, if fuel is stored in the reactor pool”	Cooling of fuel stored outside of the reactor pool is not part of the chapter on the reactor.				X	Requirement 7, SSR-3
135.	USA 30	A5.4, 3 and 4	Change parentheticals to be examples (e.g.,). This change should be considered where ever “etc.” was removed in the document.	Not sure that the parentheticals are complete lists but could be read that way by the user without the e.g.	X				
136.	USA 31	A5.4 general	Consider adding to the basic information on fuel design information on fuel qualification. Is this what para A5.9 is referring to? If it is, it is not clear.	Qualification can prove aspects of the fuel are acceptable and can be reference by the operator.	X				
137.	USA 32	A5.8	Consider adding wording on surveillance and any fuel limitations, for example, changes in length.	These surveillances are important to confirming performance within the bounds of the safety analysis.					The comment is technically correct, but surveillance requirements are addressed in other part of safety analysis report (Chapter 13).
138.	USA 33	A5.11	Consider adding wording on surveillances.	Surveillances are important to confirming performance within the bounds of the safety analysis.					See response to USA comment #32
139.	Canada 12	A.5.17	~~Most of the details described below (including hydraulic characteristics, power distribution, maximum thermal loads, nucleate boiling and flow instability) may not be required for subcritical assemblies.~~	When the word “most” is used, the reader will automatically assume they are in the category of “most” and that it is the regulator’s job to prove	X				

			For subcritical nuclear assembly facilities, ~~Most of~~ the details described below should be addressed, as applicable, commensurate with the design configuration of the specific facility.	otherwise. It is the proponent's responsibility to disposition whether detail is necessary or not based on their specific facility configuration. The guidance should be written to reflect this. The Myrrha prototype is a good example of where this may be necessary.				
Chapter 6								
140.	Canada 13	A. 6.1	~~This chapter of the safety analysis report may not be required for subcritical assemblies~~. For subcritical nuclear assembly facilities, the decision on application of this chapter should be commensurate with the safety importance of cooling systems and connected systems. A brief statement pointing to these features should be used to support the level of detail, if any, in this chapter.	When terminology such as "may not be required" is used, the reader will automatically assume they are in that category and that it is the regulator's job to prove otherwise. This style of language should be avoided particularly when use of elements of the SAR in public licensing discussions becomes more pervasive. It is agreed that the graded approach can and should be applied, but the proponent should be prompted to explain, for the specific facility why this chapter is not required based on specific features. This may be further enabled by briefly	X			

				pointing to previous information in the SAR to support the conclusion.				
141.	Japan 9	A.6.1 Last sentence	This chapter of the safety analysis report may not be required for low power research reactors, critical facilities and subcritical assemblies.	Low power research reactors and critical facilities would have no cooling system.	X			
142.	USA 34	A6.1	Consider adding wording on surveillances.	Surveillances are important to confirming performance within the bounds of the safety analysis.				Please see response to USA comment #32
143.	USA 35	A6.4	Consider adding information on monitoring radionuclides in primary coolant.	Important attribute to detect fuel elements fission product leaks and other potential problems.	X			
144.	USA 36	A6.5	Consider adding information on monitoring radionuclides in secondary coolant	Important attribute to detect heat exchanger failures and other potential problems.	X			
145.	Germany 34	A.6.8.	**Emergency core cooling system** A.6.8. The design and operation of the emergency core cooling system should be described in detail. The accident conditions for which this system is designed should be mentioned, and analyses should be provided to demonstrate that the system fulfils the requirements. The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as	The emergency core cooling system is a safety system primarily to deal with DBA. Thus, it is proposed to describe A.6.8 in chapter 7 "Engineered Safety Features".			X	Chapter 7 refers to the information on engineered safety features in other chapters of safety analysis. It is more appropriate to keep the currently recommended structure of the safety analysis report, as it was in IAEA safety standards since 1994. Changes in structure and

			drawings of the main components. The materials the components are made of should be specified, the effects of irradiation, if any, should be discussed, and any environmental effects and ageing effects should also be discussed. The procedures for inspection and testing of the emergency core cooling system should be described.					format may cause more challenges for Member States.
146.	Germany 35	A.6.9.	The design and operation of the decay heat removal system, including the ultimate heat sink, should be described in detail. ~~The accident conditions for which this system is designed should be presented and analyses should be provided to demonstrate that the system fulfils the requirements.~~ The design and performance characteristics of the main components should be tabulated. A flow and instrumentation diagram should be included, as well as drawings of the main components. The materials the components are made of should be specified; the effects of irradiation, if any, and any corrosion and ageing effects should be discussed, as well as unfavourable environmental conditions for the ultimate heat sink.	Decay heat removal in accident conditions should be described in chapter 7 “Engineered Safety Features”.			X	See response to comment Germany #34.

Chapter 7								
147.	Japan 10	A.7.1 Last sentence	Most of these features may not be required for low power research reactors, critical facilities and subcritical assemblies.	Low power research reactors and critical facilities would have no engineered safety features.		Incorporated in the text.		The text has been revised for clarity.
148.	Germany 36	A.7.1 – A.7.5		The description of chapter 7 “Engineered Safety features” should be elaborated in more detail. At least the typical safety systems and safety features for DEC should be described in this section, e.g. emergency core cooling system, diverse ultimate heat sink, diverse shutdown system, etc.)				See response to comment Germany #34.
149.	Australia 5	A.7.4	Text appears to assume that there are or will be design features specifically provided for DECs but this may not necessarily be the case for all reactors. Suggest including clarification along the lines of “where provided” or similar.	Guide should reflect the possibility of a particular reactor not requiring specific design features for DECs.	X			
150.	Australia 6	A.7.5	Again, suggest including clarification along the lines of “where provided” or similar for situation where additional safety features for DECs are not required.	See above comment on A.7.4	X			
Chapter 8								
151.	Canada 14	A. 8.9	~~The reactor power control system may not be required for subcritical assemblies.~~ For subcritical nuclear assembly facilities, the decision on application of this chapter should be commensurate with the safety importance of control systems. A brief		X			

			statement pointing to these features should be used to support the level of detail, if any, in this chapter.					
Chapter 9								
152.	Australia 7	A.9.3	Additional text again makes assumptions about design for DECs regarding an assumed need for non-permanent electrical power supplies.	See above comment on A.7.4		added "as needed"		
153.	USA 37	A9.4	Consider adding wording on surveillances.	Surveillances are important to confirming performance within the bounds of the safety analysis.				The comment is technically correct, but surveillance requirements are addressed in other part of safety analysis report (Chapter 13).
Chapter 10								
154.	Australia 8	A. 10.1	Again, suggest including clarification along the lines of "where provided" or similar for situation where additional safety features for DECs are not required.	See above comment on A.7.4	X			
155.	USA 38	A10.4, 1	Change "spent fuel" to "irradiated and spent fuel"	Many research reactors have irradiated fuel in storage that is not spent and will be returned to operation.	X			

156.	Korea 12	A.10.7. Page 71	A. 10.7. ~ A System description should also be provided. Considering the safety analyses result of design extension condition according to A.16.47-A.16.52, the habitability and good condition of control room shall be maintained in accordance with Requirement 75 of SSR-3. Additional functions of ~ in the confinement function.	The habitability of control room shall be ensured even under design extension condition to implement the procedures or guidelines for accident management.	X			
157.	Pakistan 5	A.10.8.	A description and a safety analysis of the fire protection system should be provided in this section, including information on procedures, prevention plan, **training of personnel** and maintenance activities. Reference can also be made to the design methods (see para. A.2.11).	On-site and off-site may be deleted to avoid confusion. Training of off-site personnel is question mark. Moreover, training of off-site personnel is the responsibility of town/city government instead of research reactor management.	X			
Chapter 11								
158.	Korea 13	Page 72 Para A.11.2.	A.11.2. This section should provide ~ with the research reactor. The postulated initiating events such as failure of experimental apparatus or material (e.g. loop rupture), exothermic chemical reactions, and so on (see paras. 3.22), shall be evaluated. The analysis results and the safety design features of experimental facilities with respect to these	The safety analysis for experimental facilities used for isotopes production (especially, fission moly production) is very crucial (see Annex IV–Experimental facilities). Therefore, the safety analysis results for postulated initiating events and the safety design features of the experimental facilities shall be included in Chapter 11 (Research Reactor	X			

			events shall be provided. Such facilities may include ~ should also be discussed.	Utilization) obviously.				
159.	Germany 37	A.11.5	A.11.5. Materials that will not be allowed to be used in experiments in or near the reactor core should be specified, together with materials that may be utilized only under additional safety conditions. The maximum allowable positive as well as negative reactivity of materials inserted in or near the reactor should be specified. This should include the maximum speed of insertion / withdrawal of materials.	Some probes / samples which will be irradiated will have an impact on the reactivity of the core. The allowable range of the reactivity inserted by the probe / sample need to be specified to ensure the main safety function "control of reactivity".	X			
Chapter 12								
160.	Korea 14	Page 77-78 Para A.12.26	(hyphen 1) - Locations of monitors~~ and ~~, detectors and sampler;	To detail the monitoring equipment	X			
161.	Germany 38	A.12.29 to A.12.33		Radioactive waste management should be described in a dedicated chapter "Radioactive waste management". See also our general comment No. 2.			X	Radioactive waste management is covered by chapter 12. The contents are adequately covered, the subjected comment is issue

								of format which may cause more challenges for Member States.
162.	Pakistan 6	A.12.29	Reference of IAEA requirement Documents such as IAEA GSR Part 5: Predisposal Management of Radioactive Waste may be included.	This section requires to describe the treatment of radioactive waste and for detail guidance IAEA requirement document may be mentioned as reference.	X			Added at A. 12.34
163.	Korea 15	Page 78 Para A.12.29	<u>(d) The type and size of waste container.</u>	To detail the waste process requirement	X			
164.	USA 39	A12.30(c)	Consider adding measures to ensure that effluents released to the environment are soluble.	Non-soluble material can be re-concentrated in the environment.	X			
165.	Korea 16	Page 79 Para A.12.30	(e) Requirements for the system capacity, ~ reduce leakage and prevent uncontrolled releases <u>such as overflow from tanks,</u> to the environment.	To detail the waste process requirement	X			
166.	Korea 17	Page 79 Para A.12.33	A.12.33. If applicable, ~ for explosion should be described. <u>The expected effluents concentration should be tabulated by radionuclide released, including total annual radioactive release to the environment. The dilution factors upon release should be given.</u>	To apply same requirements from liquid waste	X			

167.	Korea 18	Page 80 Para A.12.36	A.12.36. If radioactive releases have not been treated in terms of ~ a calculation of the individual doses to critical group, at the research reactor site boundary and at off-site locations, due to the effects of all releases.	To clarify the radiation exposure target	X			
Chapter 13								
168.	Finland 6	A.13.10.	These written instructions and procedures (see also Ref.NSG-4.4 [1920]) should include information on the following items, as approperiate: — Reactor startup, operation and shutdown; — Loading, unloading and movement of fuel and irradiated material; — Inspection and testing of items important to safety, in particular the safety systems; — Setting up, testing and performance of experiments with safety significance; — Maintenance, in particular concerning major components or systems important to safety; — Radiation protection; — Response to anticipated abnormal occurrences, failures of systems or components, and accident conditions; — Effluent monitoring and environmental monitoring; — Emergencies;	Please consider security including physical protection and data security. see also 4.6, 4.8	X			

			— Security including Physical protection and data security (see paras A.13.12 and A.13.13); — Fire protection. The safety analysis report should describe how to perform major, minor and temporary modifications to procedures.					
169.	USA 40	A13.10	Consider adding procedures on use of radioactive material produced in the reactor and the shipment of radioactive materials	Common activities of operating organizations and experimenters with potential safety significance.	X			
170.	Finland 7	Subtitle	**Security including Physical protection and data security** [30]			Nuclear safety and security interface		In consistence with NPP safety guide
171.	Finland 8	A.13.12.	The measures taken to protect the research reactor against unauthorized access and sabotage, and to protect against unauthorized removal of fissile and radioactive material, should be kept confidential and therefore be described in a separate plan for physical protection (see IAEA Nuclear Security Series No. 13, Nuclear Security Recommendations on Physical Protection of Nuclear Material and Nuclear Facilities (INFCIRC/225/Rev. 5) [27] and IAEA Nuclear Security Series No. 23-G [79]), including procedures for access to the site and to the research reactor, and the physical protection systems.	Please compete so that it covers physical protection and data security.		Accepted and modified		Text has been elaborated to cover safety and security interface aspects in consistence with NPP safety guide.
172.	USA 41	A13.14 general	The section is about records and reports, but there appears to be no information on reports. Considering adding a para that contains a	Reports to the regulator are an important part of conduct of operations.		The title is changed to Documents and Records as in consistence with NPP safety guide.		In coherent with NPP safety guide.

			description of reports to be made to the regulator.					
Chapter 16								
173.	France 5	A.16.1	The safety analysis presented… These analyses include deterministic safety analysis of normal operation, anticipated operational occurrences, design basis accidents and design extension conditions, ~~including~~ and analyses performed in support of 'practical elimination' of conditions arising that could lead to early radioactive releases or large radioactive releases, as well as any probabilistic safety assessment performed to complement deterministic safety analyses	See comment on 3.35 According to SSR-3, practical elimination is applicable to conditions arising (or event sequences) that could lead to early or large release. It is not applicable to the releases themselves (such an application would not be meaningful)	X			
174.	Finland 3	3.35, A.16.1. A.16.16. A.16.47. A.16.50.	Please clarify each paragraph discussing DECs with different types of acceptance criteria. DECs cover the DECs without significant core damage and DECs with melting of the reactor core. The practicle elimination concept deals with later one.			The issue has been clarified by revising the text in 2.17.		The para 2.17 has been revised to cover the acceptance criteria for DEC without significant core degradation and DEC with core melting alongwith practical elemination concept.

175.	Germany 4	A16.6	Meaning of star symbol * ?	n/a				Described in para 2 of Appendix
176.	Germany 5	A16.6	Such parameters should include, <u>but are not limited to:</u>	Other parameters might be important to certain facilities. Specifications about the reactor vessel, fuel elements, reflector, neutron source, biological shielding, or max. continuous power are given in description of research reactor facilities and could be essential for a full description	X			
177.	Germany 39	A.16.11	Each postulated initiating event should be assigned to one of the following categories, or grouped in some other manner consistent with the type of research reactor under study (some of these are not applicable to ~~sub~~critical assemblies as indicated in para 3.23):	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). Subcritical assemblies shall be deleted in this paragraph.		The text added as “for some subcritical assemblies…….”		Please also see response to Canada comment 15 and Germany comment # 7, 12, 19.
178.	Canada 15	A.16.11	Each postulated initiating event should be assigned to one of the following categories, or grouped in some other manner consistent with the type of research reactor under study ~~(some of these are not applicable to subcritical assemblies as indicated in para 3.23)~~~~:~~ For subcritical nuclear assembly facilities, areas in the categorization list provided below will be dependent on facility-	Because there can be significant variance in facility design, selection/rejection of categories should be justified in consideration of specific facility considerations.	X			

			specific design features and their importance to safety. The selection of categories and assumptions for their use should be systematically documented.					
179.	USA 42	A16.16 general	Consider adding item about engineered safety features	ESFs can be an important part of sequences	X			
180.	France 6	A.16.16	Evaluation of individual events A16.13 … A16.16 A.16.16. The step by step sequence of events, from event initiation to the final stabilized condition, should be described. The following should be provided for each event sequence: h) ~~Justification~~ for sequences that are considered as 'practically eliminated', it should be justified that they are physically impossible or extremely unlikely with a high degree of confidence.	A practically eliminated sequence does not need to be studied step by step. Consistently with SSR-3, the major concern is to demonstrate that it is physically impossible or extremely unlikely with a high degree of confidence.		(h) Justification for event sequences that are considered as 'practically eliminated' and justification that they are physically impossible or extremely unlikely with a high degree of confidence		For clarity
181.	Canada 16	A.16.22	For subcritical assemblies, parameters should and identified in consideration of facility-specific design features and their importance to safety (e.g.; measures to address reactivity accidents) ~~many of these parameters are not significantly affected by transients, and most of these parameters are not applicable (e.g. power distribution and critical heat flux ratio). For these assemblies, results of the analysis of reactivity accident considered should be presented and adequately described.~~	The list provided for research reactor are examples only. Because there can be significant variance in subcritical facility design, selection of parameters needs to be systematically performed and be in consideration of specific design features. The list provided continues to provide a suitable template for use within the systematic process.	X			

182.	Germany 40	A.16.22	(…) For ~~sub~~critical ~~assemblies~~ facilities, many of these parameters are not significantly affected by transients, and most of these parameters are not applicable (e.g. power distribution and critical heat flux ratio). For these assemblies, results of the analysis of reactivity accident considered should be presented and adequately described.	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). "subcritical assemblies" shall be " replaced by critical facilities".		The text has been modified "For critical assemblies and subcritical assemblies….		Please see response to Canada comment # 16, Germany comment# 7, 12, 19.
183.	Korea 19	Page 94 Para A.16.28	A.16.28. This section should ~ sequences (e.g. to demonstrate the effectiveness of the building or means of confinement, or to show that the resulting doses to critical groups [reference number] would meet regulatory requirements).	To add reference no. of IAEA document (BSS 115)		Corrected, GSR Part 3 is mentioned		BS115 no longer valid
184.	Germany 6	A16.31	The radionuclides released to the environment, the quantity of ~~the~~ each specific radionuclide and other physical factors characterizing the release should be given for each of the event sequences that results in releases to the reactor building.	More detailed	X			
185.	Germany 7	A16.31 e	Release mode (single puff, intermittent, continuous) and estimated release duration	Necessary for source term determination	X			
186.	USA 43	A16.32, 10	Change to "Loss of shielding (e.g., a loss of coolant accident that uncovers the reactor core but does not lead to	Add example.	X			

			cladding damage)”					
187.	Canada 17	A. 16.37	Suggest deleting due to confusing wording: ~~(The exclusion boundary is the boundary of the deliberate exclusion from the scope of regulatory control of a particular area of exposure of the research reactor on the grounds that it is not considered amenable to control by means of regulatory requirements)~~	Please reference where, in the IAEA safety framework, this definition was derived from. This definition does not make sense as written. The exclusion zone (and boundary) is typically used as either a security measure and/or to describe a potential application of the fifth level of the defence in depth and is drawn from dose acceptance criteria for normal operation, AOOs and DBAs that limit dose to critical public at the boundary exposed for a specific period of time. In many countries, the concept of an exclusion zone/boundary is specifically meant to denote an area where the licensee can exert timely and appropriate control over all activities including access by the public to prevent radiation dose. It supports timely evacuation from within the zone. It does not define a boundary of	X			

				regulatory control. Regulatory control can be exerted at any location by the regulator but is done via different legal means whether: • through the licence or (where activities are conducted illegally i.e. without a licence) the regulator has other legal means to exert regulatory controls.				
188.	Germany 8	A16.39, 11	Velocity of propagation, the distance to critical groups and the timescale over which doses are calculated.	Connected to meteorological conditions and complementary to a distance value	X			
189.	Korea 20	Page 97 Para A.16.39	A.16.39. Radiation fields associated with ~, together with estimates of doses to critical groups [reference number].	To add reference no. of IAEA document(BSS 115)		Corrected		See response to Korea comment#19
190.	France 7	A.16.47	Analysis of design extension conditions A.16.47. For design extension conditions, the results of analysis should demonstrate that the design of the research reactor is such that protective measures that are limited in terms of times and areas of application shall be sufficient for protection of the public, and sufficient time shall be available to take such measures. Moreover the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release ~~is~~ should be demonstrated as practically eliminated. The results of the analysis should confirm that protective measures that are limited in terms of time and areas of application will be sufficient for	The guidance shall be consistent with SSR-3, notably 6.68 and the corresponding objective. Moreover, practically eliminated conditions are not part of DEC		Paragraph 6.68 of SSR-3 replaced the proposed text		In consistent with SSR-3.

			protection of the public, and sufficient time will be available to take such measures. If the results of the analysis do not demonstrate meeting these criteria, additional safety features that are reasonably practicable should be implemented to prevent accident conditions beyond those considered in the design basis accident conditions, or to mitigate their consequences					
191.	Australia 9	A.16.47 to A.16.52	This section would appear to imply that the analysis of DECs should be presented in a separate section to the analysis of AOOs and DBAs but please confirm. In addition, is there any guidance on how accident sequences that are not within the design basis but are also not considered as DECs (such as large aircraft impact, seismic event significantly in excess of the design basis etc.) should be addressed?					Coherent with NPP. It is more appropriate to presented safety analysis in one chapter of SAR including DEC
192.	Australia 14	A.16.48 and A.16.51	Add statement to the effect that some DECs may deal with aspects other than fuel degradation. A.16.51 does this to a degree, but it could be mentioned further up in the section.	Refer to comment for 3.33. Again, the emphasis in this section is for fuel/core damage, and although most DECs will deal with fuel damage, there may be other DECs not related to the core or fuel.				Already covered.
Chapter 20								

193.	Germany 9	A20.1	~~However, safety precautions taken in the design and operation of the reactor may greatly reduce the possibility probability of an accident.~~	Hopefully - but accidents can and do happen	X			
194.	IRAN 1	Chapter 20: A.20.1 Second line A.20.2 First line And A.20.6 Second line A.20.5 First line A.20.6 Second line	"…emergency ~~response~~ plan…"	It is not clear why the **emergency plan** has been replaced by **emergency response plan.** The definition of emergency plan, according to GSR Part 7 and IAEA Safety Glossary is as follows which covers all the aspects of emergency preparedness and response: "A description of the objectives, policy and concept of operations for the response to an emergency and of the structure, authorities and responsibilities for a systematic, coordinated and effective response. The emergency plan serves as the basis for the development of other plans, procedures and checklists." In SSR-3, also, "Emergency plan" has been used. In subclause 3.5 of this draft, "Emergency preparedness and response plan" and in subclause 4.8, "Emergency plan" have been used. It is suggested to use	X			

				"Emergency Plan" in whole document.				
195.	IRAN 2	Chapter 20/ A.20.2/ Second line	"…on ~~accidents~~ accident conditions…"		X			
196.	IRAN 5	Chapter 20/ A.20.2	"A.20.2. This section should demonsatrate that the emergency ~~response~~ plan is coordinated with those of all other response organizations and is based on accident conditions, including… analysed in the safety analysis report.	According to the 6.19 of GSR Part 7, the emergency plan shall be coordinated with those of all other bodies that have resposibilites in a nuclear or radiological emergency. It is a key point.		This section should also demonstrate that the emergency plan is prepared in coordination with all other response organizations		A new sentence has been added.
197.	Japan 4 EPReSC	A.20.2.	This section should demonstrate that emergency plans and procedures are based on ~~conditions, including conditions that are beyond design basis accidents and conditions that are beyond design extension conditions~~ the accidents in the safety analysis report as well as those postulated for the purpose of emergency preparedness and response on the basis of the hazard assessment.	The safety analysis of NPPs is not required to identify beyond design extension conditions in the safety analysis report for the purpose of emergency preparedness and response [SSG-2]. SSR -3 requires the paragraph 7.90 in the Requirement 81.	X			
198.	Japan 7 EPReSC	A.20.1. A.20.2. A.20.6.	emergency ~~response~~ plan	In consistent with definition of GSR Part7.	X			

				"emergency plan"				
199.	Finland 9	A.20.2	This section should demonstrate that the emergency response plan is based on accidents conditions, including design extension conditions and conditions that are beyond design extension conditions, analysed in the safety analysis report.	The original text used the term "beyond design basis accident" which should not be used anymore. Question: Should the emergency response plan be based on conditions that are beyond design extension conditions as it is required here? How severe conditions?	X			The text has been revised. See also response to Iran comment no. 5 and Japan comment # 4
200.	IRAN 3	Chapter 20/ A.20.3/ Bullet (b) and (e)	"(e) ~~Notification of government authorities and local authorities~~;" Or keep bullet (e) as it is and change bullet (b) as follows: "(b) The process for identifying, and classifying and ~~notifying~~ an emergency;"	Considering the definition of "notification" in GSR Part 7, "notifying an emergency" in bullet (b) is repeated with different wording in bullet (e). GSR Part 7: "notification. (1) A report submitted promptly to a national or international authority providing details of an emergency or a possible emergency; for example, as required by the Convention on Early Notification of a Nuclear Accident. (2) A set of actions taken upon detection of emergency conditions	X			

				with the purpose of alerting all organizations with responsibility for emergency response in the event of such conditions."				
201.	IRAN 4	Chapter 20/ A.20.3/ Bullet (i)	"Arrangements with medical facilities ~~to treat contaminated individuals;~~ for medical treatment"	According to Requirement 12 (subclause 5.65) of GSR Part 7: " For facilities in categories I, II and III, arrangements shall be made to manage an adequate number of any individuals with contamination or of any individuals who have been overexposed to radiation, including arrangements for first aid, the estimation of doses, medical transport and initial medical treatment in predesignated medical facilities." So arrangements shall be made for medical treatment, individuals who have been overexposed to radiation, first aid and medical transport."	X			
202.	Canada 18	A.20.4	A.20.4. ~~Most of the details are not required~~ For low power reactors as well as critical facilities and subcritical assemblies the types and nature of details will depend~~ing~~ on the assessment of their Emergency Preparedness Category (EPC), as required in GSR Part 7 [29] and further described in Ref. [30].	Because there can be significant variance in subcritical facility design, and therefore risk profiles will vary, (for example a Keff of 0.5 will be a lower risk profile than a Keff 0.98 taking into account uncertainties) this clause should not make a categorical statement about what is required		For low power reactors as well as critical assemblies and subcritical assemblies the type and nature of details will depend on the		The wording has been modified as commented by other Member States on same para.

				or not. Instead, detail is informed by EPC therefore following a risk informed approach.		assessment of their hazard category and potential consequences of an emergency associated with the facility, as required in GSR Part 7 [29] and further described in Ref. [30].		
203.	Germany 41	A.20.4.	Most of the details are not required for low power reactors as well as critical facilities ~~and subcritical assemblies~~ depending on the assessment of their Emergency Preparedness Category (EPC), as required in GSR Part 7 [29] and further described in Ref. [30].	The hazard potential of subcritical assemblies is usually higher than for most of the research reactor (inventory several tons of fuel). Subcritical assemblies shall be deleted in this paragraph.		The text has been modified as " For low power reactors as well as critical assemblies and subcritical assemblies the type and nature of details will depend on the assessment of their hazard category and potential consequences of an emergency associated with the facility, as required in GSR Part 7 [29] and		Please see response to Canada comment #18, and Germany comment 7, 12, 19.

						further described in Ref. [30].”		
204.	UK 4	Appendix – A20.4	It may be possible to demonstrate for low power reactors, critical facilities and subcritical assemblies that many of the details identified above are not necessary or proportionate. Assessment of the Emergency Preparedness Category (EPC), as required in GSR Part 7 [29] and further described in Ref. [30] should be undertaken to determine what is required.	The current wording appears to give a “pass” to low power reactors etc for emergency preparedness without any clear test or requirement being established.		For low power reactors as well as critical assemblies and subcritical assemblies the type and nature of details will depend on the assessment of their hazard category and potential consequences of an emergency associated with the facility, as required in GSR Part 7 [29] and further described in Ref. [30].		Please see Canada comment # 18
205.	Japan 5 EPReSC	A.20.4.	Delete “Most of the details are not required for low power reactors as well as critical facilities and subcritical assemblies depending on the assessment of their Emergency Preparedness Category (EPC), a required in GSR Part 7 [29] and further described in Ref. [30].”	EPC 2&3 (e.g. low power reactors and critical facilities) are required almost the same requirements of EPC 1 according to a table A-1 “applicability of paragraphs in this publication by emergency preparedness category” of GSR Part7.		For low power reactors as well as critical assemblies and subcritical assemblies the type and nature of details will depend on the		The wording has been improved as proposed by other Member States, please see UK comment # 4, Canada comment #18

						assessment of their hazard category and potential consequences of an emergency associated with the facility, as required in GSR Part 7 [29] and further described in Ref. [30].		
206.	Japan 6, EPReSC	3.4. 3.18. A.20.6. A.20.7.	emergency ~~response~~ procedures	In consistent with definition of GSR Part7. "emergency procedures"	X			
207.	Germany 10	A20.7	The emergency response procedures should contain guidance on limits to ~~guidance~~ values for restricting exposure of emergency workers, (…)	Double entry	X			
Annex-1								
208.	USA 44	I-10 (c)	Add engineered safety features	It is important to have rules of response for ESFs	X			
Annex-II								
209.	Japan 11	II-2 / 2nd	However, input parameters related to reactivity insertion (e.g. ~~fuel cladding temperature and~~ delayed neutron	The fuel cladding temperature would not affect the reactivity insertion.	X			See also response to Canada #19

		Sentence	fraction and maximum reactivity worth for experiment) are applicable to sub critical assemblies.					
210.	Canada 19	II-2	~~Most of these items are not applicable to subcritical assemblies, depending on their design. However, input parameters related to reactivity insertion (e.g. fuel cladding temperature and delayed neutron fraction) are applicable to sub critical assemblies.~~ For subcritical assemblies, input parameters and initial conditions should be systematically identified in consideration of facility-specific design features and their importance to safety (e.g.; input parameters related to reactivity insertion – fuel cladding temperature and delayed neutron fraction) ~~many of these parameters are not significantly affected by transients, and most of these parameters are not applicable (e.g. power distribution and critical heat flux ratio). For these assemblies, results of the analysis of reactivity accident considered should be presented and adequately described.~~	The list provided for research reactors are examples only. Because there can be significant variance in subcritical facility design, selection of input parameters and initial conditions needs to be systematically performed and be in consideration of specific design features. The list provided continues to provide a suitable template for use within the systematic process.	X			See also response to Japan comment # 11
Annex-III								
211.	USA 45	III-4	Add integral burnable neutron poisons to the list	Some research reactor fuels contain burnable poisons	X			
Annex-IV								

212.	Canada 20	IV-2	IV–2. ~~Most~~ Applicability of these items ~~are not applicable~~ to subcritical assemblies will vary, depending on facility design (novelty and complexity) and hazard characteristics (potential for harm). However, the main radiation sources in subcritical assemblies are typically fuel, neutron source and sources for testing and calibration of radiation monitoring equipment.	When the word "most" is used, the reader will automatically assume they are in the category of "most" and that it is the regulator's job to prove otherwise. It is agreed that there are typical hazards in this type of facility, but the text should not automatically rule out the presence of other sources in specific cases because there can be significant variance in subcritical facility design.	X			