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: 18 November 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.	Finland 1	General	IAEA GSR Part 3 should be considered in the guide (and given as a reference).	GSR Part 3 deals with radiation safety and protection.	X			Already referred in the guide para 2.14.
2.	Germany 1	General	Please replace Terminology "~~low power reactors, critical and subcritical assemblies~~" with "research reactors with a low hazard potential" consistently in entire document. Compare paras 2.19, 2.24, 3.16, A.3.2, A.6.1, A.20.4 etc. as it is more appropriate.	Although the potential risk of research reactors is quite often associated to its thermal power, it is not always correct to use this simplification. Ensure consistency with SSR-3 para 1.9: "All the requirements established here are to be applied unless it can be justified that, for a specific research reactor, critical assembly or subcritical assembly, the application of certain requirements may be graded. Each case in which the		X "Research reactors with low potential hazards, critical assemblies and subcritical assemblies"		Consistency with SSR-3.

				application of requirements is graded shall be identified, with account taken of the nature and possible magnitude of the hazards presented by the given facility and the activities conducted.).				
3.	Pakistan (PNRA) 1	General	The recommendations in this SG may be linked with the relevant safety requirements of SSR-3 and GSR Part 1. For example if we see SSG-40, all recommendations have been linked with the relevant requirements (GSR Part 5), i.e., the requirements have been reproduced in bold text under which relevant recommendations (how to meet requirements) explained.	i. To be in line with other IAEA safety standards, e.g., SSG-40, SSG-41, SSG-47, etc. ii. Easy to understand the implementation of safety requirements. iii. Ensure consistency between the requirements and the recommendations.			X	The recommendations in the guide are already linked to the relevant requirements (which are referenced). This is consistent with the style of other safety guides.
4.	Pakistan (PNRA) 2	General	The following areas/requirements of SSR-3 need to be adequately addressed: i. Requirement 10: A**pplication of concept of defence in depth**. **ii.** Requirement 62: **Lighting systems.** **iii.** Requirement 63: **Lifting equipment.**	Missing information.		X		Application of defence in depth is addressed throughout the guide and specifically referred to in several parts of

								the main body of the guide and the Appendix (content of SAR) and Annex I also. Text added to address Req. 62 and Req. 63 in Chapter 10 of the Appendix.
Section 1								
5.	Germany 2	1.3 Lines 9-10	*The reference [3] should be updated* The same for paras 1.10 and 1.13	SSG-22 is currently under revision (working number DS511) and supposed to be published soon.		X "….SSG-22 (under revision as DS511), .."		Text added in para 1.3 where used first time and in the list of references. All references will also be updated at the final stage.
6.	Germany 3	1.4 Line 13 New footnote	[…] Furthermore, this publication covers other aspects of reactor operation normally included in the safety analysis report, such as operational limits and conditions[x], commissioning, operating procedures, and utilization and modification, which are also discussed in other publications.	Add the footnote for operating limits and conditions (the same as in the previous version of SSG-20) as it is very practicable and useful. Please keep the original	X			

			[x] The terms 'safety specifications', 'technical specifications (tech. specs) for safe operation' and 'general operating rules' are used by operating organizations and by regulatory bodies for nuclear reactors in some States instead of the term 'operational limits and conditions'. These expressions usually cover safety limits, safety system settings, limiting conditions for safe operation, surveillance requirements and administrative requirements.	text of SSG-20 or include this definition in the Glossary.				
7.	Belgium 1	1.5.	[…] this includes siting, design, ~~and~~ operation, decommissioning and dismantling of the facility.	Decommissioning and dismantling happens under regulatory control			X	See text in para 1.12. Decommissioning is out of the scope of this Safety Guide and covered in SSG-47.
8.	Germany 4	1.10 Line 2 New footnote	The recommendations provided in this Safety Guide are applicable to any type of research reactor[xx]. [xx]In this Safety Guide, the term 'research reactor' includes associated experimental facilities and subcritical and critical 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.	Add the footnote for research reactors. Footnote for research reactor (the same as in the previous version of SSG-20) may be very useful. Please keep the original text.	X			

9.	Germany 5	1.11 Line 23	[…] Homogeneous reactors and accelerator driven systems are out of the scope of this publication.	Add at the end of para 1.11 in order to provide further clarification on the scope of this guide. Sentence concerning homogeneous reactors and accelerator driven systems should be added to be conform to SSR-3.		X "…….research reactors of higher potential hazards, specialized reactors (e.g. homogeneous reactors, fast spectrum reactors) and reactors having specialized facilities (e.g. hot or cold neutron sources, high pressure and high temperature loops)……"		The text is revised for consistency with the revision of other research reactor safety guides (DS509).
10.	Russia 1/ Rostechnadzor	1.11/11	Additional recommendations on the safety analysis, on preparation of the safety analysis report and on the licensing process for research reactors with power levels in excess of several tens of megawatts, fast reactors and reactors using experimental devices such as high pressure and temperature loops and cold or hot neutron sources high powered or otherwise advanced or complex research reactors may be taken from 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][6]	Research reactors are not mentioned in the listed documents, but general approaches of these documents can be used for research reactors. The word "provided" points at the direct recommendations in the listed standards, but it is not so.	X			

11.	Russia 2/ Rostechnadzor	1.12/4	Delete the sentence "In any case, the justification for the approach selected on the basis of this Safety Guide should be provided to the regulatory body".	Operating organization can use this guide in its internal activities and it will be a "good practice".		X "In any case, the justification for the approach selected ~~on the basis of this Safety Guide~~ should be provided to the regulatory body".		For making the text more general.
12.	Germany 6	1.12	Although this Safety Guide mainly concerns newly designed and constructed research reactors, its content is applicable to any relicensing process or reassessment of a research reactor requested by the regulatory body or decided on by the operating organization. In any case, the justification for the approach selected on the basis of this Safety Guide should be provided to the regulatory body. Licensing of decommissioning activities is not discussed in detail in this Safety Guide (detailed guidance on decommissioning see SSG-47 [23])	Reference required			X	SSG-47 is already referred in relevant paras 2.48, 4.12 etc. Inclusion here will add redundancy.
13.	France 1	1.12	1.12. Although this Safety Guide mainly concerns newly designed and constructed research reactors, its content is applicable as far as reachable to any relicensing process or reassessment of a research reactor requested by the regulatory body or decided on by the operating organization. In any case, the justification for the approach selected on the basis of this Safety Guide should be provided to the regulatory body. Licensing of decommissioning activities' is not discussed in	The content might be unreachable or not reasonably practicable for older research reactors.		X "….to the extent practicable"		Consistent with SSR-3 para 4.26.

			detail in this Safety Guide.					
14.	France 2	1.14	1.14. The interfaces between nuclear safety and nuclear security should be considered in such a way that the impacts of safety on security and the impacts of security on safety are taken into account from the design stage and an appropriate compatibility is achieved. The impact of security on safety should not be taken in account in the same way than impact of hazards issued by safety analyses.	Do not confuse safety and consequences of malevolence			X	The text doesn't confuse safety and consequences of malevolence. The suggested text is not clear
15.	Germany 7	1.15-1.18	1.15. This Safety Guide addresses two interrelated issues: the safety assessment of the research reactor and the preparation of the safety analysis report. It also provides general recommendations on the conduct of the steps in the licensing of a research reactor. The main reason for presenting these two topics together in a single Safety Guide is their interrelationship and their joint importance in the licensing process. ~~1.16.~~ Section 2 describes the licensing process by which the safety of the research reactor and the issuing of licences are controlled and determined. ~~1.17~~. Section 3 presents general recommendations on the preparation of the safety analysis report, in particular the preparation of the safety analysis by the operating organization. ~~1.18.~~ Section 4 provides general recommendations on the information to be provided to the regulatory body to facilitate the process of review and assessment of the safety of the research reactor by the regulatory body.	We suggest to merge at least paras 1.16-1.18 to be consistent with other Safety Standards.	X			

Section 2								
16.	Russia 3/ Rostechnadzor	2.5/3	(1) Siting and site evaluation; (2) Design; (3) Construction; (4) Commissioning; (5) Operation, including utilization and modification; (6) Decommissioning and release from regulatory control	Design and construction are two very different stages, which can be different in time and states. Sometimes such procedure as design certification is used separately.	X			
17.	Russia 4/ Rostechnadzor	2.6/1	A detailed demonstration of nuclear safety, including an adequate safety analysis, should be submitted by the operating organization when applying for the siting and construction licenses, 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].	The first stage is siting even in the case it is accompanied by construction. If the first stage is missed, it is necessary to explain why or to start with it.		X A ~~detailed~~ demonstration of nuclear safety, including an adequate safety analysis, should be submitted by the operating organization at each stage of the licensing process ~~when applying for the construction licence~~, and should be reviewed and		See resolution for France comment 3 on para 2.6.

						assessed by the regulatory body before the next stage is authorized.		
18.	Belgium 2	2.6.	[…] important safety issues are ~~dealt with~~ identified early in such a 'prelicensing' phase.	The pre-licensing is useful to identify potential barriers to licensing. It is unlikely that those can be resolved within the limited timeframe of a pre-licensing. Milestones need to be set in the nuclear development program for resolution.		X "…….dealt with properly in the pre-licensing phase…"		Revised for consistency with SSG-12.
19.	France 3	2.6	2.6. A ~~detailed~~ demonstration of nuclear safety, including an adequate safety analysis, should be submitted by the operating organization when applying for the construction licence, and should be reviewed and assessed by the regulatory body before the next stage is authorized.	At the very first step of the applying for the construction licence, the design might not be such advanced than to enable a detailed demonstration of nuclear safety. At least, the meaning of "detailed" has to be clarified.		X A ~~detailed~~ demonstration of nuclear safety, including an adequate safety analysis, should be submitted by the operating organization at each stage of the licensing process, and should be reviewed and assessed by		Combined resolution with Russia comment 4/ Rostechnadzor on same para.e. 2.6

						the regulatory body before the next stage is authorized.		
20.	Germany 8	2.7	At all stages, the operating organization is responsible for nuclear safety and should be able to have enough financial and trained personal resources to fulfil its nuclear safety obligations and should be able to demonstrate that it has control over the research reactor and that it has an adequate organizational structure, a management system, and adequate resources to discharge its obligations and, as appropriate, its liabilities.	Addition and clarification			X	Already covered in the text of paras 2.7 and 4.2 (c).
21.	Italy 1	2.7	Page 10 Line 9 The totality of the documentation that the operating organization uses in making this demonstration, ~~some of which may not be included in the initial formal submission~~, should cover all appropriate topics, depending on the stage of the licensing process.	The last period seems to include the meaning of the previous one.	X			
22.	Russia 5/ Rostechnadzor	2.10/1	Full set of license documents including safety analysis report should be submitted to the regulatory body. Step by step submission of documents should be discussed between the operating organization and the regulatory body.	Ordinary licensing procedure starts after receiving an official statement and a set of documents by the regulatory body from the licensee. It is much better to review various sections of safety analysis report simultaneously, because they correlate to each other. In addition, the regulator should have as much as possible complete set of documents in order to plan		X "The safety analysis report or other documents with information appropriate for each stage of licensing process (see para 2.21)		Text modified for clarity.

				the work of experts.		should be submitted to the regulatory body."		
23.	Belgium 3	2.12.	Add the following: (c) Assess the organizational arrangements of the licensee to discharge its responsibilities as a responsible nuclear operator.	Organizational readiness must be checked by the regulatory authorities			X	Already covered in the text of paras 4.2 (c).
24.	Germany 9	2.12 Line 4	The operating organization should submit information to the regulatory body on the basis of which the regulatory body can determine whether the proposed research reactor can be sited, designed, constructed, commissioned, operated, utilized, modified, extended shutdown[xxx] and decommissioned without undue radiation risks to site personnel, the public or the environment. [xxx]A research reactor in extended shutdown is one that is no longer operating, with no decision on its decommissioning, and where there is no clear decision about the future of the reactor as to whether it will be brought back into operation or decommissioned. Long shutdown periods for maintenance or for implementation of refurbishment and modification projects are not considered an extended shutdown state.	An explanation or a footnote or reference to SSR-3 to clarify what is meant under "extended shutdown" would be helpful.		X placed in extended shutdown (See requirement 87 of SSR-3 [2])		Relevant SRR-3 requirement referred, and it has the footnote.
25.	Italy 2	2.12	Page 11 Line 23 This review and assessment should proceed from an overall survey of the reactor to an in-depth review and assessment of the design of individual structures, systems and components, and their behaviour in normal operation,	The word behavior seems to be more appropriate.			X	Performance is consistent with NPP guide DS449

			anticipated operational occurrences and accident conditionds					
26.	USA 8 (USNRC)	2.14, last sentence	2.14, last sentence: Requirements on radiation protection are established in IAEA Safety Standards Series No. GSR Part 3 (Rev. 1), Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [14].	For the purposes of consistency with structure used in other safety standards. Also, GSR 3 is typically referred to as the "Basic Safety Standards."	X			
27.	Belgium 4	2.15.	[…] results of the safety analysis for ~~both~~ all the operational states of the research reactor and the accident conditions considered in its design.	As written the sentence is confusing. It might be understood that 'accident conditions' are an operational state.		X "…safety analysis for ~~both~~ ~~the~~ operational states ~~of the research reactor~~ and ~~the~~ accident conditions of research reactor considered in its design "		Text is revised for clarity.
28.	Pakistan (PNRA)3	Para 2.15, bullet 3	Para 2.15, bullet 3: A set of performance requirements on systems including safety features credited in design extension conditions as defined in **Requirement 22 of** SSR-3 [2];	More clear.		X design extension conditions (see Requirement 22 of SSR-3 [2])		For consistency.
29.	France 4	2.15	2.15. Acceptance criteria should be applied to judge the acceptability of the results of the safety analysis for both the operational states of the research reactor and the accident conditions considered in its design. They may be: - A set of numerical limits on the values of predicted parameters; - A set of conditions for plant facility states	Design extension conditions should not apply for research reactors.			X	DEC is applicable to research reactors as per SSR-3 requirement 22".

			during and after an accident; - A set of performance requirements on systems ~~including safety features credited in design extension conditions as defined in SSR-3 [2];~~ - A set of requirements on the need for, and the ability to credit, actions by the operating organization, ~~including protective measures that are limited in terms of times and areas of application for design extension conditions~~.					
30.	Germany 10	2.16	~~Acceptance criteria may be specified as basic and specific. Basic acceptance criteria are aimed at achieving an adequate level of defence in depth. Examples include maximum allowable doses to the public or the prevention of fuel failure [15]. The specific acceptance criteria should include additional margins beyond the basic acceptance criteria as established within the regulatory framework, to allow for uncertainties. These specific acceptance criteria may be proposed by the operating organization and should be satisfactory to the regulatory body.~~ The acceptance criteria should include additional margins beyond the basic acceptance criteria as established within the regulatory framework, to allow for uncertainties. These specific acceptance criteria may be defined by the designer or by the operating organization. The set of acceptance criteria should be satisfactory to the regulatory body.	Text is confusing. It defines basic acceptance criteria and specific acceptance criteria, but gives recommendations only for the specific acceptance criteria. Keep the original text as in the previous version of SSG-20.			X	Guidance is provided on both basic and specific acceptance criteria in para 2.16.

31.	Israel, 1 (IAEC)	2.17	The terms *significant* (describing core degradation) and *minor* (regarding off-site radiological impact) are used. It could be useful to elaborate (in a footnote) on parameters for estimating these qualitative factors (to be agreed between the operator and the regulatory body). For example, in **paragraphs 2.44 and 4.11** of the present document, similar terms as *major safety significance* and *significant effect on safety* are refered to SSG-24 (revised as DS510B), where the method for assessing these terms are clearly pointed out.	Clarity			X	The terminologies as used are consistent with IAEA glossary and is also coherent with NPP safety guides.
32.	Pakistan (PINSTECH) 1	2.17/3-4 & 3.33/11	Footnote no. 23 on page 37 should be moved to footnote of page 12	The term cliff edge effect is mentioned first time on page 12 but its definition is given on page 37	X			
33.	France 5	2.17	~~2.17. 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 margins to avoid cliff edge effects and there is no, or only minor, off-site radiological impact. In accordance with para 6.68 of SSR-3[2], the conditions arising that could lead to an early radioactive release or large radioactive releases should be practically eliminated and 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. The analysis may lead to~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			~~implementation of additional safety features, or extension of the capability of safety systems to maintain the main safety functions and to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material confined in the facility, including radioactive material resulting from the degradation of the reactor core.~~					
34.	USA 1 (USNRC)	2.18(b), 5th bullet	Limits for significant damage to fuel and fuel cladding failures	Correct grammar and generalize the concept of fuel damage	X			
35.	Germany 11	2.18 (a) Line 3	[…] (a) Radiological criteria such as: — Maximum allowable doses to the public; — Dose limits (or design target doses[xxxx]) for site personnel; …… [xxxx]Guidance on design target doses is provided in INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors, IAEA Safety Standards Series No. NS-G-4.6, IAEA, Vienna (2008). paras 2.8 and 2.9.	Add the footnote for design target doses. Footnote for design target doses (the same as in the previous version of SSG-20) may be very useful. Please keep the original text.		X (or design target doses, see NS-G-4.6 […..])		
36.	Germany 12	2.19	Where specific acceptance criteria mentioned above are determined may not to be applicable to ~~low power~~ research reactors with a low hazard potential, critical assemblies and subcritical assemblies, the specific acceptance criteria should be justified and documented. Additionally, for subcritical assemblies, there may be acceptance criteria specified for limits	According to SSR-3 para 2.17 and SSG-22 para 2.7 thermal power is only one factor to characterize the hazards originating from a research reactor. The more general term "hazard potential" is more suited.		X "Research reactors with low potential hazards…."		See resolution to Germany comment 1

			on insertion of reactivity that prevent criticality.					
37.	USA 2 (USNRC)	2.19	Where specific acceptance criteria mentioned above are determined ~~may~~ not to be applicable to low power research reactors, critical assemblies and subcritical assemblies, the specific acceptance criteria should be justified and documented. Additionally, for subcritical assemblies, there may be acceptance criteria specified for limits on insertion of reactivity that prevent criticality.	Correct typographical error	X			
38.	Belgium 5	2.19	Where specific acceptance criteria mentioned above ~~are determined~~ may not ~~to~~ be applicable to low power research reactors, critical assemblies and subcritical assemblies, the specific acceptance criteria should be justified and documented.	Sentence seems incorrectly structured.		X		See resolution to USA comment 2.
39.	Belgium 6	2.20.	Remove the entire paragraph from this Section	This paragraph as written is not about 'acceptance criteria' but requirements to comply with the principle of defense-in-depth.			X	The paragraph is about acceptance criteria and it is appropriate to retain the original text.
40.	Belgium 7	2.20 (first dash)	… a design basis accident by itself should not generate a design extension condition~~s~~.	If the paragraph would be kept, either "a design extension condition" or "design extension conditions"	X			
41.	France 6	2.20	2.20. The acceptance criteria should include the following: - An event should not generate a more serious condition of the research reactor without the occurrence of a further independent failure. Thus	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			an anticipated operational occurrence by itself should not generate a design basis accident, a design basis accident by itself should not generate a ~~design extension conditions~~beyond design basis accident. - There should be no consequential loss of function of the safety systems necessary to mitigate the consequences of an accident. - Systems used for mitigation of the consequences of accidents should be designed and constructed depending on their importance to safety, to withstand the maximum loads and stresses and the most extreme environmental conditions for the accident analysed.					
42.	Russia 6/ Rostechnadzor	2.22/8	The procedure for submitting the safety analysis report and other documents at each stage is determined by the regulator.	The licensing procedure is determined by national government/regulator and operating organization should follow the established procedure and format.	X			The comment is correct but not relevant to the paragraph 2.22 which address the preparation of safety analysis report.
43.	Germany 13	2.23 and 2.24	2.23. At various stages in the course of the design process (for example, before the start of construction or operation), the status of the design should be described in the safety analysis report, and the description should include the design and safety assessment that has been carried out up to that point. ~~2.24~~. For research reactors with low potential hazard~~, particularly critical assemblies and subcritical assemblies,~~ the amount of information and analysis to be provided according to paras. 2.26 and 2.48 could be	Para 2.24 gives no new recommendation. Please combine with para 2.23 (don't make a separate point). Additionally, there exist also critical and/or subcritical assemblies with higher potential hazard – we made suggestions for the text		X "For research reactors with low potential hazards, critical and subcritical assemblies…."		The text is retained as separate para. It is consistent with SSR-3. Also see resolution to Germany comment 1 and 12.

			reduced in accordance with a graded approach.					
44.	Italy 3	2.24	Page 15 Line 3 ……..amount of information and analysis to be provided according to paras 2.26 and 2.48 could be reduced or increased in accordance with the graded approach.	The period is completed.			X	The paragraph specifically refers to reactors with low potential hazards.
45.	Russia 7/ Rostechnadzor	2.25/1	Since the approval of one licensing stage is normally required before commencement of the next stage, the safety analysis report should be made available for review and assessment on a timescale that has been determined by the regulatory body. Regulator estimates of the size and scope of the analyses. The established period should be reasonable to perform independent review of the submitted documents at a certain stage.	According Requirement 25 of GSR Part 1 (Rev. 1): Review and assessment of information relevant to safety: The regulatory body shall review and assess relevant information whether submitted by the authorized party or the vendor, compiled by the regulatory body, or obtained from elsewhere to determine whether facilities and activities comply with regulatory requirements and the conditions specified in the authorization. This review and assessment of information shall be performed prior to authorization and once again within the lifetime of the facility or the duration of the activity, as specified in regulations promulgated by the regulatory body or in the			X	Para 2.25 is not in conflict with the regulatory authority to establish schedule of submission. Requirement 25 GSR Part 1 doesn't address the schedule of submission. No text suggested by the comment for para 2.25. It is adequate as is.

				authorization. So, the regulator is responsible for timing within the licensing procedure.				
46.	Belgium 8	2.28.	Somewhere in the text around paragraph 2.28 a new paragraph needs to be introduced with text along these lines "For innovative designs that make use of new fuels and coolants, additional safety margins and monitoring requirements need to be introduced by the designer in order to cope with the lack of operating experience. This applies as well to the results of safety analysis performed using deterministic codes that might have been developed for other type of reactors and qualified using experimental data obtained using other types of fuel and coolants."	Research reactors based on Gen IV concepts need to be taken into account.			X	The scope of the document is governed by SSR-3 which does not include "Gen IV concepts".
47.	Italy 4	2.29	Page 17 Line 1 To obtain a construction licence ~~or an approval~~ for the start of construction……	It is proposed to remove the reference to "approval" as in case of authorization for construction the term "licence" appears more appropriate			X	In some jurisdictions the construction licence is not given but approval is required. The text is also consistent with SSG-12 para 3.40 where approvals are also needed.
48.	Russia 8/ Rostechnadzor	2.30/1	Those aspects of the design that should be submitted to the regulatory body for review and assessment before the design is finalized should be	The set of safety assessment materials submitted to the regulator should be sufficient		X "…..in		

			identified in agreement with the regulator so that activities can proceed while the reactor is under construction. The information should be updated and resubmitted to the regulatory body as the detailed design and the construction of the reactor proceed. In some cases, revised versions of documents will be sufficient; in other cases, technical supplements may be appropriate. Additional guidance on the licensing process for this stage is given in SSG-12 [10].	to start the review and so should be determined by regulator according to SSG-12, item 2.13. Procedures for issuing authorizations for each stage of the lifetime of the installation and for each type of installation should be prepared by the regulatory body, to ensure that all necessary steps have been taken prior to the granting of a license.		agreement with the regulatory body…..”		
49.	Russia 9/ Rostechnadzor	2.31/1	The safety analysis report is the main document provided at these stages for review and assessment by the regulatory body for the authorization of the detailed design and/or construction.	According to SSG-12 design and construction are different stages and separate authorizations can be given for design and construction.		X “…detailed design and for construction”		The text is revised for clarity.
50.	Pakistan (PNRA) 4	2.32	..... The updated safety analysis report should be resubmitted to the regulatory body for review and assessment in order to obtain the required authorization for commissioning.	The statement seems not in line with SSG-12. Please provide the reference to support the statement.			X	The statement is based on paragraph 3.4 of SSR-3.
51.	Italy 5	2.33	Page 18 line 7 The test results should be approved by the operating organization at the appropriate level of management and, as necessary depending on the relevance of the safety tests, by the regulatory body before the subsequent test sequence is started.	Some tests and the relative results has to be approved by the regulatory body		“….depending on the direct affect on safety”		As per SSR-3 para 7.51.
52.	Italy 6	2.33	Page 18 Line 13 — Stage A: tests prior to fuel loading (pre operational tests);	For completeness			X	SSR-3 requirement 73 is quoted as it

								is and additional text cannot be added.
53.	Belgium 9	2.34.	Delete the entire sentence “Such tests and measurements … subcritical assemblies”.	It is not right to suggest to use the results of commissioning tests to validate the computational models and tools used to design the object being commissioned.		X Such tests and measurements should be used to ~~validate~~ verify the results from computational models and tools that are used for design and safety analysis of the subcritical assemblies		The text is retained for additional guidance.
54.	Germany 14	2.34	For subcritical assemblies, initial criticality tests and low power tests of Stage B and tests of Stage C are not applicable. However, tests should be performed to verify that the configuration is subcritical. Some other tests, ~~such as approach to criticality and~~ e.g. 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.	Deleted text is in contradiction with SSR-3 para 6.62. The design of subcritical assemblies shall include technical provisions to prevent criticality (see para. 6.66). Perhaps „subcriticality verification test“ is meant here?		X Some other tests such as verification of adequate subcriticality ~~such as approach to criticality~~ and neutron flux measurements are also needed.		Text is modified for clarity.

55.	Italy 7	2.37	Page 19 Line 19 If deviations from the design documentation have occurred, they should be recorded, assessed and it should be shown that the safety analysis has not been remains valid, otherwise a new safety analysis has to be conducted.	For completeness		……it should be shown that the safety analysis remains valid and that safety has not been compromised"		Consistent with SSG-12 para 3.50
56.	Russia (SEC NRS 1)	2.37	Edit item Stage A should ensure that the reactor has been constructed, manufactured and installed correctly and in accordance with the design documentation. *The stage should include such tests, like checks of reactivity control system performance, verification of research reactor alarm systems, tests of neutron flux control systems and other necessary tests.* If deviations from the design documentation have occurred, they should be recorded, assessed and it should be shown that the safety analysis remains valid. The results of this stage should also confirm the operational features of the research reactor and should lead to the development procedures, which should be confirmed during Stages B and C.	It is proposed to include exact tests of research reactor main equipment and systems that need to be implemented on stage A (prior to fuel loading).			X	Although the additional text proposed is valid however the level of detail is not appropriate here.
57.	Italy 8	2.38	Page 19 Line 7 ……the operational limits and conditions for commissioning, and the preliminary operating procedures [17], ~~and~~ the interface between nuclear safety and security should be taken into account at this commissioning stage	Editorial	X			
58.	Russia 10/ Rostechnadzor	2.38/1	Stage B is an important step in the authorization process.	At this step it is usually no fission products in the core and so it is very important step but term "major" is strict in view of translation		X Stage B is a significant step in the authorization		Consistent with SSG-12 para 3.45

				and is often translated in other languages as "upmost".		process.		
59.	Russia 11/ Rostechnadzor	2.39/1	As power Stage C moves closer to completion <u>core moves closer to equilibrium nuclear fuel burnup</u>, this commissioning stage should focus on how the research reactor will be operated, utilized and maintained, and on procedures for controlling and monitoring operation and for responding to deviations and other occurrences. Before authorization for routine operation is requested, the test results, any corrections of non-conformances, modifications to the design or modifications to the operational procedures, and any proposed changes to the operational limits and conditions should be submitted to the regulatory body for review and assessment.	Approach to equilibrium nuclear fuel burnup is different from normal operation in view of fuel loading procedure and values of criticality parameters. Therefore it would be good to mention this period.			X	It is not always the case that the core moves closer to equilibrium nuclear fuel burnup at Stage C of commissioning - unnecessary detail, not appropriate to add it.
60.	Russia 12/ Rostechnadzor	2.42/17	If required as a result of the review of the safety measures for operation, the operating organization should submit to the regulatory body a request for an amendment of the license or license conditions, which should include a revised safety analysis report.	Review itself is a part of safety assessment and so its results should be included in the SAR and therefore submitted to the regulator.		X If required as a result of the review of the safety measures for operation, the operating organization should submit to the regulatory body a request for an amendment of the license, which ~~may~~ <u>should</u> include a revised safety analysis report,		For clarity.

						as appropriate.		
61.	Pakistan (PNRA) 5	2.42	2.42. The safety analysis report should be ~~prepared~~ **updated** for the operating licence application.	In line with para 3.57(e) of SSG-12.	X			
62.	Germany 15	2.43 Line 9 The same for paras 2.44, 4.11 and A.11.1	*The reference [6] should be updated*	SSG-24 is currently under revision (working number DS510B) and supposed to be published soon.		X "….SSG-24 (under revision as DS510B), .."		Text added in where used first time and in the list of references. All references will also be updated at the final stage.
63.	Belgium 10	2.44.	"Experiments and modifications having major safety significance should be categorized …"	The categorization criteria used for this purpose should be somewhat clarified.		X Further guidance on utilization and modification, including categorization is provided in SSG-24 [6].		Reference to SSG-24 is already provided.
64.	Russia 13/ Rostechnadzor	2.48/1	Further guidance on decommissioning is provided in the IAEA Safety Standards Series No. Ref.SSG-47, Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [23].	As pointed above in item 2.47 and in item 7.22 of SSG-47 A safety assessment is a key supporting document to the final decommissioning plan. The licensee is required to prepare this document and submit it for	X			No change made to text.

				review to the regulatory body in accordance with national regulatory framework. The scope of the safety assessment, its content and the degree of detail may vary depending on the complexity and hazard potential of the facility. Safety assessment is still important in view of radiation hazards, but the term 'SAR' is not used at the decommissioning stage. Fuel should be removed before the decommissioning stage.				
Section 3								
65.	Russia 14/ Rostechnadzor	3.1/1	To meet applicable safety requirements on the safety analysis report including requirements 1, 5 and 6 provided in SSR-3 [2], the operating organization should make arrangements for preparing a safety analysis report to demonstrate the safety of the research reactor.	SAR also includes safety justification or operating procedures, but not the design only.			X	Agree that the SAR covers design and operation but operation is addressed in the second sentence.
66.	Belgium 11	3.3	… updated to describe: — … — ~~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 (for	It is excessive to describe events that may have occurred during the lifetime of the research reactor, especially those at other nuclear installations.		X "Changes in the facility and its operation (as a result of		Text modified for clarity.

			instance as a result of operating experience feedback or events that happened) and that may influence the actions that will need to be taken during decommissioning of the research reactor.			events that may have occurred during the lifetime of the research reactor or operating experience feedback, including from other nuclear installations) and that may influence the actions….”		
67.	Russia 15/ Rostechnadzor	3.4/1	Reference to para 3.6, Requirement 1 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.	It is proposed to make corrections in both documents. SAR should include full description of the research reactor, otherwise, it will be impossible for regulator to assess correctness of operating organization classification of the systems. In real practice, regulator opinion on safety significance of some systems can differ from the applicant opinion.			X	Retain for consistency with SSR-3 requirement 1.

68.	Belgium 12	3.5.	Modify the sentence as follows: "The safety analysis report should also provide details of the internal emergency plan, and design principles related to decommissioning ~~plan~~".	The external emergency plan is set by the regulator together with the federal and local authorities. The decommissioning plan is not normally included in the SAR. The SAR might contain design principles related to decommissioning though (relative for example to limit of sizes and availability of space for decommissioning operations, the minimization of the use of materials that could be activated…), but that is not a decommissioning plan.		X "details of the emergency plan, decommissioning plan and design principles related to decommissioning"		Using the term "emergency plan" is more appropriate.
69.	Israel 2, (IAEC)	3.6	Regarding separate documents to which reference can be made in the Safety Analysis Report (instead being part of the Report itself) we suggest to add a few words (in a footnote) emphasizing that these separate documents have to be formally completed before submitting the SAR, in order to serve as references to the SAR. The same issue is mentioned also in paragraph 3.11.	Clarity			X	It is understood that reference documents to SAR needs to be formally completed.
70.	Belgium 13	3.10	… Guidance on public participation is given in SSG-12 [10].	Dot (.) is missing at the end.	X			
71.	Germany 16	3.16	For ~~low risk~~ facilities with a low hazard potential ~~(such as some critical assemblies, subcritical assemblies, or research reactors with~~	The potential risk of the facility is related not only to its thermal power.		X "(such as some critical		See resolution to Germany comment 1and

			~~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 low potential hazard facilities.	SSR-3 2.17 defines criteria for a grading. They do not to be specified here. Furthermore, there exist critical and/or subcritical assemblies with significant potential risk.		assemblies, subcritical assemblies, or research reactors with a low potential hazards,"		12
72.	Belgium 14	3.18	… and the development of the accident management procedures and the …	There will be more than one accident management procedure	X			
73.	Pakistan (PNRA) 6	Para 3.19	3.19. The consideration of fault conditions should determine …….. ……	The terminology 'fault condition' mentioned in this para need to be defined as it is not found in IAEA Safety Glossary. Furthermore, this terminology is not used in SSR-3.			X	This is commonly understood term and is used in NPP guide on same topic (DSS449)
74.	Belgium 15	3.20 (first dash).	- That sufficient defence in depth has been provided, and that the levels of defence are preserved and balanced to the extent possible in that potential accident sequences are arrested as early as possible.	No undue burden should be placed on a level of defence in depth compared to the others (e.g. increased mitigation by sacrificing prevention).		"…..are independent and preserved"		Consistence with SSR-3.
75.	Belgium 16	3.20 (first dash)	- That sufficient defence in depth has been provided, ~~and~~ that the levels of defence are preserved to the extent possible, and ~~in~~ that potential accident sequences are arrested as early as possible.	Sentence seems incorrectly structured	X			

76.	Germany 17	3.22 Bullet (2)	[…] (2) Insertion of excess reactivity: — Criticality during fuel handling and loading (e.g. due to an error in fuel insertion, dropping of fuel assembly on core); — Startup accident; — Control rod failure or control rod follower failure; — Control drive failure or control drive system failure; — Failure of other reactivity control devices (such as a moderator or reflector); — Unbalanced rod positions; — Failure or collapse of structural components; — Insertion of cold or hot water; — Changes in the moderator material (e.g. voids or leakage of D2O into H2O systems or leakage of H2O into D2O system); — Effects of experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material, error in loading or unloading); — Insufficient shutdown reactivity margin; — Inadvertent ejection of control rods; — Maintenance errors with reactivity devices; — Spurious control system signals; — Removal of poisons from the coolant or moderator. — <u>Incident of control rod;</u> — <u>Dropping of fuel elements onto moderator tank;</u> — <u>Control rod with reduced absorption effects</u>	Addition			X	Partially covered in 3.22 (2). List of PIEs are as established in SSR-3.

77.	Germany 18	3.22 Bullet (3)	[…] (3) Loss of flow: — Primary pump failure; — Reduction in flow of primary coolant (e.g. due to valve failure or a blockage in piping or a heat exchanger); — Effect of the failure or mishandling of an experiment; — Rupture of the primary coolant boundary leading to a loss of flow; — Fuel channel blockage or flow reduction (e.g. due to foreign material); — Improper power distribution due to, for example, unbalanced rod positions, in core experiments or in fuel loading (power-flow mismatch); — Reduction in coolant flow due to bypassing of the core; — Deviation of system pressure from specified limits; — Loss of heat sink (e.g. due to the failure of a valve or a pump, or a system rupture). — Converter plate cooling failure; — Moderator cooling failure	Addition			X	Partially covered in 3.22 (2). List of PIEs are as established in SSR-3.
78.	Germany 19	3.22 Bullet (4)	[…] (4) Loss of coolant: — Rupture of the primary coolant boundary; — Damaged pool; — Pump down of the pool; — Failure of beam tubes or other penetrations.	Addition			X	List of PIEs are as established in SSR-3

			— Leakage at separating bulkhead between reactor pool and fuel pool					
79.	Belgium 17	3.22 (4) (third dash)	Loss of coolant: … — Pump down of the pool;	A "failure" mechanism is missing (as for the other 3 dashes: rupture? damage? failure?)			X	Consistence with SSR-3
80.	Belgium 18	3.22 (5) (last dash)	— Exceeding of fuel ratings.	Is this an initiating event on its own? Is it not a consequence of other initiating events? To be deleted?			X	List of PIEs are as established in SSR-3.
81.	Belgium 19	3.22 (8)	— Extreme meteorological phenomena;	The dash is missing in front	X			
82.	USA 3 (USNRC)	3.22 (7)	-Hurricanes; -snow and ice storms;	Hurricanes and snow storms should be separate items in the list			X	List of PIEs are as established in SSR-3.
83.	France 7	3.28	3.28. The safety analysis should identify design basis accidents ~~and design extension conditions without significant fuel degradation and design extension conditions with melting of the reactor core~~. In addition, accidents beyond the design basis that have more severe consequences may should also be analysed for purposes of emergency planning and for specifying the measures to be taken to mitigate the consequences of an accident.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
84.	France 8	3.29	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 are characterized by conservatism and are based on defined sets of rules for event selection, analytical	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			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 (arising from the design basis accident or from a 'maximum credible accident') 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 engineering judgement, without the benefit of determining the probabilities of the event sequences.						
85.	Germany 20	3.30	Deterministic and probabilistic approaches ~~are~~ should be used to supplement each other in the abovementioned safety assessments.	Clarification		X		Reverted back to original text for consistency with SSR-3.
86.	Germany 21	3.31 Line 28 New Issue	[…] A deterministically derived list of design extension conditions without significant fuel degradation should be developed. The relevant design extension conditions should include: (a) Initiating events that could lead to situations beyond the capability of safety systems that are designed for design basis accidents. (b) Anticipated operational occurrences or frequent design basis accidents combined with multiple failures that prevent the safety systems	It is not clear how to deal with DEC. We suggest to use formulation from SSG-2, Rev1, paras 3.40 and 3.45			X	DEC are addressed in paragraph 3.33 to 3.37.

			from performing their intended function to control the postulated initiating event. The failures of supporting systems are implicitly included among the causes of failure of safety systems. The identification of these sequences should result from a systematic analysis of the effects on the plant of a total failure of any safety system credited in the safety analysis, for each anticipated operational occurrence or design basis accident (and in particular for the most frequent, anticipated operational occurrences and design basis accidents). (c) Credible postulated initiating events involving multiple failures causing the loss of a safety system while this system is used to fulfil its function as part of normal operation. The identification of these sequences should result from a systematic analysis of the effects on the plant of a total failure of any safety system used in normal operation. A number of specific sequences with core melting (severe accidents) should be selected for analysis in order to establish the design basis for the safety features for mitigating the consequences of such accidents, in accordance with the plant safety objectives. These sequences should be selected in order to represent all of the main physical phenomena (e.g. reactor decay heat or containment status) involved in core melt sequences.					

87.	Germany 22	3.33 Line 2	[…] Design extension conditions include events more severe than design basis accidents that originate from extreme events or combination of them that could cause damage to the SSCs important to safety or challenges the fulfillment of main safety functions as well as progressions of events that could lead to reactor core damage or ~~some other~~ radiological release.	Clarification	X			
88.	France 9	3.33	~~3.33. Requirement 22 of SSR-3[2] provides requirements on design extension conditions. Design extension conditions include events more severe than design basis accidents that originate from extreme events or combination of them that could cause damage to the SSCs important to safety or challenges the fulfillment of main safety functions as well as progressions of events that could lead to reactor core damage or some other radiological release. The examples of design extension conditions that are applicable to research reactors can be found in Ref [24]. The analysis of design extension conditions should be performed with best estimate codes, models, initial and boundary conditions to demonstrate that core melting can be prevented or mitigated with an adequate level of confidence and there are adequate margins to avoid any cliff edge effects[23].~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
89.	France 10	3.34	~~3.34. Analysis of design extension conditions, including assessment of the response of the research reactor to those conditions, should demonstrate that the design of the facility is adequate to prevent accident conditions or to mitigate their consequences as far as reasonably practicable. The results of analysis may require additional safety features for design extension~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			~~conditions or extension of the capability of safety systems, to fulfill the main safety functions and to ensure the capability for managing accident conditions in which there is a significant amount of radioactive material confined in the facility, including radioactive material resulting from the degradation of the reactor core~~					
90.		3.35	~~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 and;~~ ~~Control locations remain habitable to allow performance of required actions.~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
91.	Belgium 20	3.35.	A reference should be given to a document providing the definition of (or more explanation on) 'practical elimination'	It is not always obvious what 'practical elimination' means.			X	'practically eliminated' is defined in the glossary and need not be defined here.
92.	France 12	3.36	~~3.36. The analysis should address the impact of the most challenging conditions and demonstrate that the compliance with acceptance criteria is achieved by safety features implemented in the design, combined with the implementation of procedures or guidelines for accident management.~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
93.	Belgium 21	3.37.	" … likelihood of criticality shall be sufficiently remote to be considered ~~as a design extension condition~~ practically eliminated."	The risk of criticality should be practically eliminated for subcritical assemblies.			X	SSR-3 requirement quoted.

94.	France 13	3.37	~~3.37. 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 ensure adequate margins to avoid any cliff edge effects as well as for identification of additional safety features, or extension of the capabilities of safety systems, to prevent or mitigate the consequences of such event.~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
Section 4								
95.	Russia 16/ Rostechnadzor	4.1/13	The review and assessment process include inspections on the site and elsewhere to verify the claims made in the submissions.	For common wording with SSG-12.	X			
96.	Russia 17/ Rostechnadzor	4.2 (f) /1	Determine whether the operational limits and conditions are arranged under the regulatory requirements and whether an adequate level of operational safety can be ensured, including the provisions made for accident conditions;	Usually, requirements determine procedure for safety limits and conditions determination and does not provide the specific values.		X "….are established in accordance with the regulatory requirements ….”		For clarity.
97.	Belgium 22	4.2 (g)	Determine whether the utilization and modification of the research reactor meet the requirements of the regulatory body.	This phrase looks confusing and difficult to interpret. Strictly speaking the regulatory body does not have ‘requirements’ for the utilization of the research reactor. The license will state what can be done with the reactor and the regulatory body will verify	X			No text proposed.

				(by inspections) that the operator stays at all time within the operating limits and conditions.				
98.	Germany 23	4.2 Line 2	[…] The regulatory body can then determine whether the proposed research reactor can be sited, constructed, commissioned, operated, utilized and modified, and ~~eventually~~ decommissioned, without undue radiation risks to site personnel, the public or the environment.	Clarification	X			
99.	Germany 24	4.7	The documents of the operating organization's case for the safety of the research reactor as presented in the safety analysis report, which should be submitted to the regulatory body for review and assessment in order to obtain authorization for commissioning Stage A (tests prior to fuel loading), should include: (a) The 'as-built' design of the reactor; (b) The <u>non-nuclear</u> commissioning programme; (c) The operational limits and conditions for Stage A commissioning; (d) The records and reporting systems; (e) The management system, organizational structure and programme for operation.	Clarification (non-nuclear)		X The commissioning programme for Stage A		Stage A as used in the text refers to non-nuclear commissioning.
100.	Germany 25	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	Clarification (nuclear)		X The revisions to the commissioning programme for Stage B, if any;		Stages are defined in the text.

			commissioning stage, including non-conformances and, where appropriate, their associated corrective actions; (b) The revisions to the nuclear commissioning programme, if any; (c) The operational limits and conditions for Stage B commissioning; (d) The provisions for radiological protection; ……					
101.	Indonesia 1	4.10	(Adding new item) (f) The ageing management programme	The ageing management programme determined at design phase should be revised according to the latest condition of structure, system and component after construction and commissioning phases.		X		The comment is valid. However, there is no need to include it as the list is not exhaustive. Ageing management aspects are already covered throughout the SAR. This further highlighted in the revised text of Chapter 13 of Appendix.
102.	Pakistan (PNRA) 7	4.10	4.10. …… ……. (e) The arrangements for maintenance, periodic	The bolded text is added to make it in line with para 3.57 of SSG-12.		X 4.10 …… (f)		Updates to documents is already

			testing, inspection, control of modifications and changes to specifications and surveillance**;** **(f) Arrangements for emergency preparedness and response;** **(g) Updated documents as mentioned under para 4.6 above.**			Arrangements for emergency preparedness and response;		covered in para 4.10 (d).
103.	Germany 26	4.12	Before the authorization for ~~eventual~~ decommissioning and release from regulatory control can be obtained, the application submitted to the regulatory body for review and assessment should include:…..	Clarification	X			
104.	Russia 18/ Rostechnadzor	4.12/5	—The decommissioning plan.	As main document.	X			
105.	Russia 19/ Rostechnadzor	42.12	Add the third hyphen: —The results of decommissioning plan realization.			X “Before the authorization for decommissioning ~~and release from regulatory control can be obtained,~~ the application submitted to the regulatory body for review and assessment should		The text is modified for clarity.

						include: — The records and results of operational experience; — The decommissioning programme. Before release from regulatory control can be obtained, the results of decommissioning should be submitted to the regulatory body. Detailed guidance on decommissioning is provided in		

						SSG-47 [23].		
Appendix								
106.	USA 4 (USNRC)	Appendix, 1st paragraph	The amount of information and the level of detail may vary depending upon the type, complexity and the design of the facility.	Typographical error	X			
107.	USA 5 (USNRC)	Appendix, 2nd paragraph	In addition, some technical contents of those mentioned in this Appendix may not be applicable to some types of ~~the~~ subcritical assemblies. ~~Contents that may not be applicable to some of the subcritical assemblies~~ These are highlighted throughout the Appendix by an asterisk (*), or specifically indicated.	Grammar and reduce repetition	X			
108.	Pakistan (PNRA) 8	Appendix	Following information may be added: i. Chapter 11 may also include the description of "**Experimental Devices**". **ii.** Chapter 18 may also include description of **(1) Management of resources and (2) Measurement, assessment and improvement of the management system.** iii. Chapter 19 may also include description of **Optimizing of the facility's layout and access routes** iv. Radioactive Waste Management (RWM) is a separate area from Radiation Protection so a separate chapter may be introduced for RWM.	i. Please see Requirement No. 66 of SSR-3. (ii) Please see paras 4.13, 4.15 and 4.20 of SSR-3 iii. Please see para 6.92(b) of SSR-3. iv. To be in line with IAEA safety standards.			X	The text adequately address items (i, ii and iii). There is no need to highlight specifically those aspects. (iv) RWM is covered in the contents of SAR. It is more appropriate to keep the current format

								as it was in IAEA safety standards since 1994. Changes in the format may cause more challenges for Member States.
109.	Germany 27	Appendix Line 13	[…] ~~The information required for the content of the safety analysis report for subcritical assemblies should be the same as for research reactors. However, with the application of graded approach, the amount of information and the level of detail should be consistent with the complexity and lower hazards of subcritical assemblies. In addition, some technical contents of those mentioned in this Appendix may not be applicable to some types of the subcritical assemblies. Contents that may not be applicable to some of the subcritical assemblies are highlighted throughout the Appendix by an asterisk (*), or specifically indicated~~.	Information is misleading. There exist also critical or subcritical assemblies with higher potential hazard. Sufficient advice for using graded approach is given already in the first paragraph ("The amount of information and the level of detail may vary depending upon the type, complexity and the design of facility."). Please remove also all asterisks (*) in the Appendix and Annexes, because they are not adequate. E.g. some of marked issues could also be omitted for other reactors (e.g Provisions for safely storing a sufficient number			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

				of spent fuel elements (A2.4), reactor system pressure (A16.6)). Justification of grading shall always be done based on the facility specification.				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 also be revision to SSG-22 on use of graded approach.
Chapter 1								
110.	Germany 28	A.1.2	In this section, a summary of the principal characteristics of the research reactor and the site should be provided. The general arrangement and layout of the research reactor should be described, starting with the core and continuing with the secondary and tertiary systems and the reactor building, to convey an impression of the research reactor and its systems, structures and components, important to safety.	Clarification (SSC)	X			
Chapter 2								

111.	France 14	A.2.2.	A.2.2. This section should describe the safety objectives and the general design requirements followed in the design of the research reactor, in consideration of the requirements for normal operation, anticipated operational occurrences, design basis accidents ~~and design extension conditions~~ and the accidents taken into account in the design. Safety objectives and design requirements for prevention of accidents and mitigation of consequences should also be included. Other measures that can be used to mitigate accident conditions should be described in the appropriate chapters of the safety analysis report.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
112.	USA 6 (USNRC)	A.2.3(r)	"Interface" should be "interfaces" or "the interface"	grammar	X			
113.	France 15	A.2.3.	A.2.3. A statement of the overall safety objectives should be included. This should be followed by a brief description of the underlying safety objectives and general design requirements that are important to the design. Safety objectives are discussed in section 2 of SSR-3 [2], and general design requirements are discussed in section 6 (see Requirements 16–41) of SSR-3 [2]. These objectives and requirements may include the following: (a) Management system requirements; (b) High standard of engineering design and, in particular, conservative design margins, engineered safety systems (features), barriers to radionuclide transfer and protection of these barriers; (c) Inherent safety features (those relying only on physical properties); (d) Passive safety features (passive features do	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			not actively change state); (e) The extent to which unique or unusual features that may affect the consequences or the probability of releases are incorporated; (f) The extent to which redundancy, diversity, physical separation and functional independence are applied in the design of safety systems and engineered safety features, so as to achieve necessary reliability of these systems and features and to protect against common cause failures; (g) Fail-safe features; (h) Defence in depth applied in the design, including the independent effectiveness of the different levels of defence; (i) Accident prevention; (j) Accident management; (k) Proven engineering practice and use of generally accepted standards; (l) Assessment of human factors and dependent failures; (m) Radiation protection; (n) Provisions for utilization and modification; (o) Provisions for ageing management; (p) ~~Features for design extension conditions~~; (q) Provisions for emergency preparedness and response; (r) Provisions for interface between nuclear safety and nuclear security.					
114.	Germany 29	A.2.3 Line 33	[…] (l) Assessment of human factors and dependent failures; (m) Radiation protection; (n) Provisions for utilization and modification;	Addition		X		Already covered in para A.2.11 and A. 2.4 (15).

			(o) Provisions for ageing management; (p) Features for design extension conditions; (q) Provisions for emergency preparedness and response; (r) Provisions for interface between nuclear safety and nuclear security. (s) Fire Protection (t) Environmental Protection					
115.	Indonesia 2	A. 2.3 (p)	(p) Features for design extension conditions (if design extension conditions exist based on the result of safety analysis);	For low power research reactors, design extension conditions may not exist due to their extremely small probability of occurrences.			X	Design extension conditions need to be considered.
116.	Germany 30	A.2.4 (22) (a)	[…] Provisions for safely storing a sufficient number of spent fuel elements and irradiated core components; *	Please remove asterisk (*) Not clear why this shall not be applicable to subcritical assemblies. While critical (sub-)assemblies may not need to change fuel during their lifetime, there is no reason why provisions from irradiated core components are excluded.		X Provisions for safely storing a sufficient number of spent fuel elements* and irradiated core components;		Critical and subcritical assemblies may not need to change the fuel.
117.	France 16	A.2.4.	A.2.4. The specific design requirements applied should be stated in this section. These requirements are discussed in detail in section 6 (see Requirements 42–66) of SSR-3 [2] and include: (1) Management system requirements for design, including codes of practice utilized in design. (2) Monitoring of variables and control of reactor and system variables within their operating ranges.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			(3) Reactor core integrity requirements. (4) Protection against flow instabilities and suppression of power oscillations*. (5) Criteria for sharing of common structures, systems and components important to safety between facilities at the same site (e.g. emergency power supply, on-site fire brigade). (6) Consideration of human factors and ergonomic principles to reduce the potential for human error and to relieve stress for the operating personnel. (7) Requirements for design analysis with validated techniques, models or codes. (8) Provision of reactivity control, including: (a) Redundant reactivity control*; (b) Reactivity limits; (c) Availability of sufficient negative reactivity to maintain the reactor subcritical under all operational states and accident conditions. (9) Design of reactor coolant system and related systems, including: (a) Requirements for adequate core cooling for all operational states and accident conditions*; (b) Requirements for coolant system integrity and protection of the boundary from leakage*; (c) Preventing the uncovering of the core*. (10) Reactor core and fuel design, including: (a) Fuel design bases for neutronic, thermohydraulic, mechanical, material chemical design; (b) Safety margins for fuel design parameters*; (c) Verification of fuel integrity; (d) Prevention of inadvertent fuel movement;				

			(e) Design bases for mechanical, thermal and chemical design of reactor materials important to safety; (f) Shutdown margins*; (g) Prevention of criticality for subcritical assemblies. (11) Provisions for safe utilization and modification, including: (a) Radiation protection for all operational conditions; (b) Design requirements to ensure that safety system settings are not adversely affected; (c) Provisions to preserve the means of confinement and shielding of the reactor; (d) Recognition of the interdependence between the reactor and any installed experimental equipment. (12) Reactor safety systems including: (a) Provision of systems for shutdown, fuel cooling and control of radionuclide releases; (b) Operating requirements; (c) Separation requirements for safety system and control functions; (d) Single failure criteria; (e) Fail-safe mode requirements. (13) Reliability and testability of instrumentation and control systems, including: (a) Provision of means to achieve required level of reliability; (b) Periodic testability; (c) Fail safe characteristics; (d) Functional diversity.; (14) Capability for surveillance and maintenance of safety related equipment. (15) Radiation protection in systems including: (					

			(a) Control of radioactive releases; (b) Stationary dose rate meters for monitoring at places routinely accessible and at suitable locations in anticipated operational occurrences and accident conditions; (c) Monitors and laboratories for determining the concentration of selected radionuclides; (d) Monitors and control of effluents; (e) Equipment for measuring radioactive surface contamination, doses to and contamination of personnel; (f) Monitoring at gates and other entrances; (g) Arrangements to assess the impact in the vicinity of the facility. (16) Buildings and structures including: (a) Building and structures designed for design basis accidents ~~and, as far as practicable, for design extension conditions*~~; (b) Requirements for leaktightness of the reactor building and the ventilation system. (17)…					
118.	USA 7 (USNRC)	A.2.4(8)	"Provisions for…"	grammar	X			
119.	Germany 31	A.2.5	The basis for safety classifications and the list of classes should be presented in this section of the safety analysis report. Additional guidance is presented in IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [26]. The approach 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 classifications and the list of classes should be presented in this	Clarification	X			

			section of the safety analysis report.					
120.	France 17	A.2.13.	A.2.13. This section should describe the scope of the qualification programme and the qualification procedures adopted to confirm that the items important to safety, ~~including safety features for design extension conditions~~, are capable of meeting the design requirements and of remaining fit for purpose in the range of individual or combined environmental challenges identified for the situations under which they are supposed to perform. The identified challenges should take into account all the stages and their duration in the lifetime of the research reactor facility.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
Chapter 3								
121.	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 some ~~low power~~ research reactors with a low hazard potential, critical assemblies 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 some of the subcritical assemblies.~~	Sufficient advice for using graded approach is included in the second sentence ("For some low power research reactors, critical assemblies and subcritical assemblies which present very limited hazards, the amount of detail provided in this chapter can be substantially reduced."). Further information are only misleading. Additionally: According to SSR-3 2.17 and SSG-22 2.7		X "For some research reactors with low potential hazards,…."		See resolution to Germany comment 1 and 12. The text is retained as it provides additional guidance.

				thermal power is only factor to characterize the hazards originating from a research reactor. The more general term "hazard potential" is more suited.				
122.	France 18	A.3.7.	A.3.7. This section should describe the appropriate methods adopted for establishing the external effects that will constitute the postulated initiating events for important natural phenomena and human induced effects. Attention should be paid to the external hazards that could potentially lead to common cause failures of the safety systems ~~and additional safety features for design extension conditions.~~ Further information on design criteria for protection against these effects should be given in Chapter 2 of the safety analysis report (see para. A.2.6).	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
123.	Germany 33	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 some low hazard, low power reactors, critical assemblies and some of 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.~~ This section should also cover all aspects of site activity that have the potential to affect the radiological impacts of the site throughout the lifetime of the reactor, including construction, operation under normal conditions and decommissioning	Compare with para. 3.16. The radiological aspect may be even more important for low risk facilities than for high risk facilities.			X	The context of para 3.6 and A.3.17 is different.

Chapter 4								
124.	Pakistan (PNRA) 9	A.4.5.	The design and operation of cranes or other lifting and **handling** devices should be described.	The bold text "handling" devices should also be added. The handling tools (underwater and other) are important to prevent dropping of objects in the pool.	X			
Chapter 5								
125.	Belgium 23	A.5.4.	(h) Fuel operating experience (if any)	Add this point to the text.	X			
126.	Belgium 24	Before A.5.13.	~~Nuclear~~ Core design	Given the content of paragraphs A.5.13-16 the use of the word 'core' seems more appropriate.			X	The paragraph coveres more than core design and it is also consistent with NPP DSS449
127.	Belgium 25	A.5.17.	"… will be available to keep the reactor fuel thermal parameters within acceptable levels ~~in a thermally safe condition~~, and that ~~an~~ adequate ~~thermal~~ safety margins will be maintained …"	'conditions' would need to be defined somewhere in the text.	X			
128.	Germany 34	A.5.24 (c)	[…] The reactor pool or tank and related components constituting the primary coolant boundary;	Most of the research reactors have only a pool. There are only few research reactors worldwide that have a reactor tank.	X			

Chapter 6								
129.	Germany 35	A.6.1 Line 6	[…] For low ~~power~~ hazard research reactors, critical assemblies and subcritical assemblies, this chapter of the safety analysis report should be commensurate with the safety importance of cooling system and connected system.	According to SSR-3 2.17 and SSG-22 2.7 thermal power is only factor to characterize the hazards originating from a research reactor. The more general term "hazard potential" is more suited.		X "For research reactors with low potential hazard…."		See resolution to Germany comment 1 and 12.
130.	Finland 3	A.6.5	Please add: The chemistry data for the secondary coolant should be presented.	This data may be safety relevant (e.g. corrosion aspect).			X	The intention of the statement is already covered (corrosion control measures and ageing effects).
131.	Belgium 26	Chapter 6 and in particular A.6.1	A "description" (see A.6.1) of the systems is not sufficient. For all systems described in this chapter the safety functions delivered by those systems need to be listed and described in the SAR.	General remark on Chapter 6			X	The text shows that the contents of the 'description', which covers the safety functions, performance characteristics, design requirements, etc.

132.	Belgium 27	Chapter 6	Moderator System paragraph (A.6.7) needs to be removed.	'The design and operation of the moderator system' sounds strange. To our knowledge the moderator that is introduced in nuclear reactors to thermalize the spectrum is not a 'system'. Or is a moderator "system" specific for some designs?			X	The moderator system paragraph should be kept, to cover all reactor designs.
133.	Belgium 28	Chapter 6	Emergency core cooling system does not belong in this chapter	ECCS belongs in Chapter 7 'Engineered Safety Features', unless the system performs functions in normal operation too, e.g. during start up or shut down.			X	Although the ECCS is an engineered safety feature it is more appropriate to keep it in this Chapter, for user benefit preserving the The structure of the safety analysis report, as it was in IAEA safety standards since 1994.
134.	Pakistan (PNRA)10	A.6.8.	A.6.8. ……. The procedures for inspection and testing of the emergency core cooling (ECC) system should be ~~described~~ **mentioned**.	The text should be modified as proposed. Because there is no need to describe the procedure in detail, instead the Procedure No. should only be mentioned.	X			

135.	Belgium 29	Before A.6.10.	Primary ~~purification~~ conditioning system	The system normally does more than purifying. It conditions the coolant so that it remains as little aggressive as possible for the core and other internals.			X	It is purification
Chapter 7								
136.	France 19	A.7.1.	A.7.1. This chapter of the safety analysis report should identify and provide a summary of the types, locations and functions of the engineered safety features provided in the research reactor for anticipated operational occurrences and accident conditions. Examples of engineered safety features are an emergency core cooling system and a containment system or a means of confinement. The requirements of these systems and supplementary features are discussed in Requirement 43 and Requirement 48 of Ref.SSR-3 [2]. ~~Examples of safety features for design extension conditions are additional cooling water supply and non-permanent equipment e.g. portable diesel generator~~. For low power research reactor, critical assemblies and subcritical assemblies, the decision on application of this chapter should be commensurate with the safety importance of engineered safety features. A brief statement pointing to these features should be used to support the level of detail, if any, in this chapter.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
137.	Germany 36	A.7.1 Line 9	[…] For low ~~power~~ hazard research reactors, critical assemblies and subcritical assemblies, the decision on application of this chapter should be commensurate with the safety importance of engineered safety features.	According to SSR-3 2.17 and SSG-22 2.7 thermal power is only factor to characterize the hazards originating from a research reactor. The more general		X "For research reactors with low potential hazards…."		See resolution to Germany comment 1 and 12 etc..

				term "hazard potential" is more suited.				
138.	France 20	A.7.4.	~~A.7.4. The design specifications of safety features for design extension conditions, where provided, should be described, along with a description of the capability of these features for preventing or mitigating the radiological consequences, including their reliability to the functions that they are required to fulfill, independence from those used in more frequent accidents, and capability of performing in the environmental conditions pertaining to design extension conditions.~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
139.	France 21	A.7.5.	A.7.5. Reference should be made to the relevant chapters of the safety analysis report, where provided, or to other documents where the engineered safety features ~~and additional safety features for design extension conditions~~ are further described further.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
140.	Belgium 30	A.7.5.	"Reference should made to …"	Typo (insert blank between "made" and "to")	X			
Chapter 8								
141.	Pakistan (PNRA) 11	A.8.1	A.8.1. ……. The requirements for instrumentation and control systems are established in paras **(6.167–6.171),** Requirement 49 of SSR-3	Relevant paras number added.		X The requirements for instrumentation and control systems are established in ~~paras~~ Requirement 49 of SSR-3		SSR-3 requirement is mentioned instead of paras.

Chapter 9								
142.	France 22	A.9.3.	A.9.3. This section should describe the design and operation of the emergency power supply, ~~including provisions for non-permanent equipment necessary to restore the electrical power supply in design extension conditions, as needed~~, and should emphasize the connection to the off-site power supply.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
Chapter 10								
143.	France 23	A.10.1.	A.10.1. This chapter of the safety analysis report should provide information concerning the auxiliary systems included in the research reactor. The description of each system, the design bases for the system and for critical components, a safety assessment demonstrating how the system satisfies the requirements of the design basis, information on the testing and inspection to be performed to verify the capability and dependability of the system, and information on the instrumentation and control system required should be provided. ~~The storage system for non-permanent equipment used in design extension conditions, where applicable, should be described.~~ In cases where auxiliary systems are not related to the protection of the public against exposure to radiation, enough information should be provided to allow understanding of the design and function of the auxiliary system; emphasis should be placed on those aspects that might affect the research reactor and its safety features or that might contribute to the control of radioactive material inside the research reactor. For those systems, foreseeable ageing effects that could affect safety should also	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			be discussed.					
144.	Germany 37	A.10.1 Line 8	[…] ~~The storage system for non-permanent equipment used in design extension conditions, where applicable, should be described.~~	Design provisions to control DEC should be addressed in chapter 7 of the SAR.			X	It is more appropriate to retain it in this chapter 10, consistent with NPP guide (DSS449)
145.	Pakistan (PAEC) 1	A.10.5	……. demineralized water. In each case, the information provided should include the design bases, a system description, flow and instrumentation diagrams, a safety assessment if required, testing and inspection requirements, instrumentation requirements and foreseeable ageing effects.	Addition/modification in the text (underlined) is proposed to account for the ageing effects that could affect the desired function of the system.	X			
146.	Pakistan (PAEC) 2	A.10.6	…….The discussions should include the design bases, a system description, a safety assessment, testing and inspection requirements, instrumentation requirements and foreseeable ageing effects.	Addition/modification in the text (underlined) is proposed to account for the ageing effects that could affect the desired function of the system.	X			
147.	Pakistan (PAEC) 3	A.10.7	The systems for heating, ventilation air and conditioning systems provided for all areas of the reactor building should be discussed in this section. This discussion should include design basis, a system description, testing and inspection requirements, instrumentation requirements and foreseeable ageing effects. Consideration should be given to the safety analyses result of design extension condition according to A.16.47-A.16.52, maintain the	The text may be modified to account for design basis, testing and inspection requirements and ageing effects.	X			

			habitability and good condition of control room in accordance with Requirement 75 of SSR-3……					
148.	France 24	A.10.7.	A.10.7. The systems for heating, air conditioning and ventilation provided for all areas of the reactor building should be discussed in this section. ~~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 should be maintained in accordance with Requirement 75 of SSR-3~~. A system description should also be provided. Additional functions of ventilation systems may be discussed in other relevant chapters of the safety analysis report, for example, ventilation systems used in the confinement function	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
149.	Finland 2	A. 10.7	Why are the systems of the reactor building excluded? Is there any specific para for discussion of them?	There should a specific place for discussion of ventilation systems of the reactor building.	X			Covered in the reactor building Chapter 4 (para A. 4.1 to A. 4.7)
150.	Pakistan (PAEC) 4	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, <u>suppression and control plan,</u> training of personnel and maintenance activities	The text is added (underlined) to account for important phenomenon of fire suppression and control.	X			
Chapter 11								
151.	Pakistan (PAEC) 5	A. 11.2	…….directly or indirectly with the research reactor. Such facilities may include the beam tubes, the thermal column, in-core or	Test is added (underlined) to include common experimental	X			

			moderator facilities, boreholes, pneumatic rabbit system and experimental loops.…	setup i.e. rabbit system.				
152.	Pakistan (PAEC) 6	A. 11.6	The maximum allowable positive as well as negative reactivity of materials used in the experiments inserted in or near the reactor should be specified. This should include the maximum speed of insertion / withdrawal of materials.	Test is added (underlined) as it is the materials that may affect the reactivity.	X			
Chapter 12								
153.	Pakistan (PAEC) 7	A. 12.1	This chapter of the safety analysis report should describe, for normal operational conditions: (a) The radiation…… ……. …. (d) Waste Management Programme and waste management system … .	Test is modified to include requirement of waste management program required by GSR Part 5.	X			
154.	Pakistan (PAEC) 8	A.12.11	……. — Control of contamination of personnel and equipment; — Methods and procedures for decontamination of personnel and equipment; — Control of compliance with applicable regulations for the transport of radioactive material; — The methods and procedures for	Text is modified to account for the decontamination of personnel and equipment. Further, to separate health surveillance from individual exposure assessment.			X	The list is not meant to be exhaustive. The items are covered in the text and further expansion is not necessary.

			personnel monitoring, including methods for recording, reporting and analysing results; — The programme for assessment of internal radiation exposure, such as bioassay or whole body counting; — The programme for health surveillance of radiation workers and other related medical surveillance of personnel, in particular in cases of overexposure; — …… — ……					
155.	Pakistan (PAEC) 9	A.12.15	This section should give a brief description of the radiation protection training programme for the management and staff responsible for radiation protection, operations and maintenance and for other personnel, including contractors and experimenters/students	Text is modified to include relevant groups i.e. operators and maintenance of research reactor. Experimenters may not be students.	X			
156.	Pakistan (PAEC) 10	A.12.18	All normal potential radiation sources (contained sources liquid and airborne radioactive material) due to reactor operation and all potential radiation sources throughout the research reactor……	The term liquid radioactive material is one of the significant radiation source.	X			
157.	Pakistan (PAEC) 11	A.12.29	This section should describe the minimization and treatment of solid radioactive waste including, as applicable: (a) The types and class of radioactive waste, the origins and quantities of solid radioactive waste, including the physical form, volume and isotopic compositions, and the	Concept of waste minimization is added in the proposed text.	X			

			measured or estimated activity; (b) For wet radioactive waste, the methods of dehydration; (c) The methods of collection, segregation, processing, packaging, storage and transport of radioactive waste; (d) The type and size of …….					
158.	Pakistan (PAEC) 12	A.12.33	A. 12.33 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. A.12.34. If applicable, design provisions to handle hazardous gases with a potential for explosion should be described.	This para is split in two paras as the content is dependent.	X			
159.	Finland 5	A.12.37	Please replace "the individual doses to critical group" with "the dose to the representative person of the public".	The member of the critical group has been replaced with the representative person (GSR Part 3).	X			
160.	USA 9 (USNRC)	A.12.6 Line 5	personnel monitoring (e.g. ~~film badges,~~ thermoluminescence dosimetry services).	Recommend removing film badges as an example since it is an outdated technology not used much for personnel monitoring.	X			
Chapter 13								
161.	Pakistan (PAEC) 13	A.13.10	These written instructions and procedures….. …..	Response to BDBA is included in the text.			X	BDBA is superseded by DEC.

			…… - Response to anticipated abnormal occurrences, failures of systems or components, and accident conditions; - Response to beyond design base accidents (BDBA)/severe accidents;					
162.	Russia 20/ Rostechnadzor	A.13.11	This section should describe the conduct of the maintenance, aging, periodic testing and inspection programme for equipment and components of the research reactor, which should be based on the guidance provided in Ref.NS-G-4.2 [30].	Description of aging program of operating organization should be included in SAR because it is not the same as maintenance, periodic testing and inspection but connected with the mentioned procedures.	X			Text added in Chapter 13 on ageing management programme
Chapter 16								
163.	France 25	A.16.1.	A.16.1. The safety analysis presented in this chapter forms the focal point of the safety analysis report. In previous chapters, it is stated that the research reactor design, and especially the design of structures, systems and components important to safety, should be evaluated for the susceptibility of structures, systems and components to malfunctions and failure. In this chapter, the effects of anticipated process disturbances and postulated component failures and human errors (postulated initiating events) should be described, including their consequences, to evaluate the ability of the research reactor to control or to accommodate such situations and failures. These analyses include deterministic safety analysis of normal operation, anticipated operational occurrences, design basis accidents ~~and design~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			~~extension conditions~~ without significant fuel degradation ~~and for design extension conditions with core melt~~ 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.					
164.	France 26	A.16.4.	A.16.4. This section should provide a brief summary, under the following headings: (1) Methods of identification, selection and justification of postulated initiating events. ~~(2)~~ Classification of the postulated initiating events in anticipated operational occurrences, design basis accidents ~~and design extension conditions~~ ~~(3)~~(2) Methods of analysis, including where appropriate: (a) Event sequence analysis; (b) Transient analysis; (c) Evaluation of external events and special internal events; (d) Qualitative analysis; (e) Radiological consequence analysis. (4) Acceptance criteria.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
165.	Germany 38	A.16.6.	A summary should be given of the research reactor parameters and ranges for specified operating conditions considered in the safety analysis. Although these values may be tabulated in various other sections of the safety analysis report, they should be summarized here to assist in the review and assessment of the safety analysis. Such parameters should include, but are not limited to:	Clarification	X			

			(a) Core power; (b) Core inlet temperature; (c) Fuel element cladding temperature; (d) Reactor system pressure*; (e) Core coolant flow*rate; ……..					
166.	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 (for some subcritical assemblies, the categorization will be dependent on facility specific design features and their importance to safety. The selection of groups or categories and assumptions for their use should be systematically documented): (a) Loss of electric power supplies; (b) Insertion of excess reactivity; (c) Loss of flow; (d) Loss of coolant; (e) Erroneous handling or failure of equipment; (f) Special internal events including failure of experiments; (g) External events; (h) Human error. (i) Loss of moderator	Addition			X	Loss of moderator is not a category - it is covered under A.16.11 (b) (See para 3.22 (2))
167.	France 27	A.16.12.	A.16.12. The basis for the categorization and grouping of postulated initiating events should be described and justified. The list of scenarios to be addressed in this chapter of the safety analysis	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			report should cover anticipated operational occurrences, design basis accidents ~~and design extension conditions~~. The postulated initiating events in each group should be evaluated to identify the events that would be bounding, and the events selected for further analysis should be indicated and justified. The events selected for further analysis should include those having potential consequences that are bounding for all other postulated initiating events in the group. 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: (a) Identification of significant occurrences on a timescale, for example, neutron flux monitor trip or start of insertion of control rods; (b) Indication of the proper functioning of normally operating reactor instrumentation and controls, and of their failure to function; (c) Indication of proper functioning of reactor protection, safety systems and other engineered safety features, and of their failure to function; ~~(d) For design extension conditions, additional failures that were assumed in the event sequence;~~ (d) Indication of the required operator actions; (e) Evaluation of dependent failures and human errors; (f) Qualitative evaluation of sequence probabilities (if employed);					
168.	Belgium 31	A.16.13.	(g) Barriers being threatened; (h) Physical and/or chemical phenomena threatening the barrier.	To be added to the list			X	This is, among others, covered by event

								sequence, transient analysis and classification of damaged states.
169.	France 28	A.16.20.	A.16.20. This section should describe the computational models employed, including computer codes or analogue simulations used in the analyses. The description should demonstrate that the models are applicable for the expected range of operational parameters, that they represent all important physical phenomena and that they have been properly verified and validated. Description should also demonstrate that computational models use conservative approaches in the case of anticipated operational occurrences and design basis accidents~~, and best estimate approaches in the case of design extension conditions~~. This section should provide only a summary of mathematical models and computer codes or lists used, referring to detailed descriptions in documents available to the regulatory body.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
170.	Russia 21/ Rostechnadzor	A.16.20/3	The description should demonstrate that the models are applicable for the expected range of operational parameters, that they represent all important physical phenomena and that they have been properly verified.	Computational models including geometry and materials cannot be validated in view of novelty of research reactor design and absence of benchmark experimental data. It is better to speak about validation of computer codes for the initial data which correlate with the code validation results.	X			

171.	Russia 22/ Rostechnadzor	A.16.20c /1	A summary of results of verification and validation studies, including:	See A.16.20	X			
172.	Russia 23/ Rostechnadzor	A.16.20c /2	*Add the new item "i".* Description of used validation models.	At the stages of design, construction and commissioning experimental data on particular research reactor are not available and other most applicable experimental models are used for codes validation.	X			
173.	Germany 40	A.16.22 last paragraph	~~For critical assemblies and subcritical assemblies,~~ ~~p~~ Parameters should be identified depending upon the facility design features and their importance to safety. (e.g. measures to address reactivity accidents).	This statement is true for all research reactors not only for critical and subcritical facilities.		"For research reactors with low potential hazards, critical assemblies and subcritical assemblies…."		For consistency.
174.	Belgium 32	A.16.31.	(g) Energy of the release;	This is an important parameters to characterize the release typically used as input for the calculations.			X	This level of detail is not provided in NPP guidance.
175.	Finland 4	A.16.37	Please replace "the most highly exposed member of the public" with "the representative person of the public"	The member of the critical group has been replaced with the representative person (GSR Part 3).	X			
176.	France 29	A.16.47. A16.48. A16.49.	~~A.16.47. In SSR-3 [2], para 6.68 states that "the design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated. The design shall be such that for design extension conditions,~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			~~protective measures that are limited in terms of time and areas of application shall be sufficient for protection of the public, and sufficient time shall 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.~~ ~~A.16.48. This section of the safety analysis report should present the assumptions used and the results obtained from the analysis of design extension conditions without significant fuel degradation. The analysis should demonstrate with an adequate level of confidence that core melting can be prevented and that there are adequate margins to avoid any cliff edge effects.~~ ~~A.16.49. This section should also present the assumption used and results obtained from the analysis of design extension conditions with core melting with subsequent releases of radioactive material to the containment (or to the research reactor building).~~					
177.	France 30	A16.50. A16.51. A.16.52.	~~A.16.50. This section should also provide identification of the most severe parameters resulting from core melt sequences, and should demonstrate that:~~ ~~• The research reactor can be brought into the state where the confinement function can be maintained in the long term;~~ ~~The structures, systems and components of the research reactor are capable of avoiding any early radioactive release or large~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			~~radioactiverelease;~~ ~~• Compliance with the acceptance criteria is achieved by features implemented in the design, combined with the implementation procedures or guidelines for accident management;~~ ~~• The possibility of conditions arising that could lead to an early radioactive release or large radioactive release is practically eliminated.~~ ~~A.16.51. This section should also describe the analysis of additional accidents, e.g. large release of tritiated heavy water, damage of targets, that are postulated for the purposes of emergency preparedness and response.~~ ~~A.16.52. The scope and content of the information provided for design extension conditions should be similar to that described above for design basis accidents~~					
178.	France 31	A.16.55.	~~A.16.55. For design extension conditions, the results of the analysis should demonstrate that the criteria defined in paragraph 2.17–2.20 have been met.~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
Chapter 17								
179.	Indonesia 3	A.17.6	— Core configurations and design limitations (e.g. reactivity coefficients, radial and axial power peaking factors, burnup limits, minimum and maximum number of fuel elements, their geometrical arrangements, inspection);	Radial and axial power peaking factors are important parameters in identifying the hotspot position in the core and preventing the boiling crisis phenomena which threaten the integrity of		X Core configurations and design limitations (e.g. reactivity coefficients, power peaking		The list is not meant to be exhaustive and only selected examples are included.

				fuel cladding.		factors, burnup limits…)		
Chapter 18								
180.	Pakistan (PAEC) 14	A.18.4	This section should describe or should refer to the particular parts of the management system that have been established for the phases of design, procurement, construction, commissioning, operation and decommissioning, as appropriate……..	One of the important phase of research reactor "decommissioning" is included	X			
Chapter 20								
181.	Germany 41	A.20.4	For low ~~power~~ hazard research 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 [34] and further described in Ref. [35].	According to SSR-3 2.17 and SSG-22 2.7 thermal power is only factor to characterize the hazards originating from a research reactor. The more general term "hazard potential" is more suited.		X "For research reactors with low potential hazards…."		See resolution to Germany comment 1
182.	Pakistan (PAEC) 15	A.20.4	…….an emergency associated with the facility, as required in GSR Part 7 [34] and further described in IAEA EPR-Research Reactor, Generic Procedures for Response to a Nuclear or Radiological Emergency at Research Reactors, EPR-Research Reactor [35].	To make it consistent with other sections of the standards.			X	IAEA style of writing references.
Annex-1								

183.	Belgium 33	I-13	" … that can lead to many similar event sequences and that may have a low cumulative ~~probability~~ frequency." " … cumulative ~~probability~~ frequency of similar initiating events … ."	Event sequences (and certainly initiating events) are characterized by an occurrence frequency, not by a probability. This comment might apply to (several?) other text strings in the document.	X			
184.	France 32	I-6	I–6. Certain methods can be used to group postulated initiating events as follows: (a) Postulated initiating events that require similar safety functions, which determine the design parameters of the safety systems; ~~(b) Postulated initiating events that require similar safety functions, which determine the parameters of the additional safety features for design extension conditions;~~ (c)Postulated initiating events that have a similar influence on reactor behaviour or on structures, systems or components, for which similar calculational models are used; (d)Postulated initiating events that can assist in the selection of limiting cases for analysis in each group; (e)External postulated initiating events that have the potential for a common cause impact on the research reactor. One possible grouping is shown in para A.16.11 of the Appendix to this publication.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
185.	France 33	I-8	I–8. A clearly defined method will facilitate the evaluation of the step by step sequence of events, from the initiation of the event to the final stabilized condition. The rules or conventions regarding the extent to which research reactor systems, including the reactor protection system	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			~~as well as additional safety features for design extension conditions~~, are assumed to function are the basis for this method. If there is a possibility of fuel cladding failure, then other barriers to prevent the spread of radioactive material have to be considered, not only if all systems function correctly but also if some of them fail. Consideration has to be given to the types of event that will be evaluated by using this method, and the types of event that will be evaluated by other methods (see paras I–15–I–19).					
186.	France 34	I-9	I–9. The sequences have to include the response of the reactor core, the research reactor systems, engineered safety features and safety features as well as human interactions. Possible sequences for the case in which a system fails need to be described in detail for accident conditions. The following points need to be considered: (a) Use of structured techniques, such as event trees or event sequence diagrams; (b) Identification of significant occurrences on a timescale, for example, neutron flux monitor trip and start of insertion of control rods; (c) Indication of correct and incorrect functioning of normally operating reactor instrumentation and controls; ~~(d) Additional failures assumed for safety features for design extension conditions;~~ (e) Evaluation of the three principal main safety functions (shutting down the reactor, cooling the fuel and maintaining confinement of radioactive material), including an indication of both the correct functioning of reactor protection and safety systems ~~as well as safety features for design extension conditions~~ and their possible failure;	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.

			(f) Required Credited operator actions for functioning of manually operated safety systems for design basis accidents and safety features for design extension conditions; ~~(g) Credited protective measures for design extension conditions;~~ (h) Frequency or probability evaluations to be carried out in assessing the sequence of events …					
187.	France 35	I-10	I–10. Rules or conventions have to be established to determine the response of reactor systems. These rules or conventions need to refer to: (a) The effect of single, random failures; (b) System qualification (or lack of qualification) under accident conditions; (c) Safety and protection systems engineered safety features ~~as well as safety features for design extension conditions~~, including reliability in quantitative terms, if applicable; (d) Support systems, such as normal and emergency electric power and for cooling; (e) Redundant trip parameters; (f) Actions of systems that are independent; (g) Operator action (e.g. response time, display of information on a console); (h) The effect of failures assumed for safety systems for design basis accidents ~~or safety features for design extension conditions~~; (i) Carrying out of frequency or probability evaluations to assess the system response, the extent to which such evaluations will be used and the methods to be employed (including validation).	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
188.	France 36	I-13	I–13. The frequency or probability of event sequences may be evaluated; this would help to determine which sequences could be excluded from the design basis ~~and considered under~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4

			~~design extension conditions~~ or to assess the relative risk presented by various sequences. This evaluation includes:…					on para 2.15.
189.	France 37	I-17	~~I-17. For events, design extension conditions have to be specified for a range of frequency of occurrence for which additional design features have to be provided to maintain the main safety functions, especially the confinement function.~~	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
190.	France 38	I-20	I–20. The significant results of the safety analysis for anticipated operational occurrences, design basis accidents ~~and design extension conditions~~ and the comparison with the acceptance criteria (see paras 2.14–2.18 20 of this Safety Guide) are presented in the safety analysis report.	Design extension conditions should not apply for research reactors.			X	See resolution for France comment No. 4 on para 2.15.
Annex-II								
No comment								
Annex-III								
191.	Belgium 34	Annex III	This Annex needs to be completely re-thought: • It could be necessary to add something to III-1 to take into account ADS. • Things are repeated – Almost all the items for which a detailed description is asked in III-2 are already mentioned in III-1 (where a 'brief' description is asked). • In III-3 the reactor protection system is also important (not only 'mechanical' and 'electrical' but I&C as well). • In III-3 the total worth of the control system and of individual rods is an	It is recommended to re-think Annex III and to re-write it from scratch.		X "~~Reactivity control system~~ <u>reactivity control mechanisms</u>"		The contents of the Annex remain useful. ADS out of the scope, reactivity control is different than shutdown system, control rod worth

			important parameter. • The last section on 'REACTIVITY CONTROL SYSTEMS' could be merged with 'REACTIVITY CONTROL SYSTEM, REACTOR SHUTDOWN SYSTEM'					changes with core configuration.
192.	Pakistan (PNRA)12	Annex-III	Each para of Annex-III may be referred at suitable place in the text of APPENDIX.	For example, reference of Para III-1 of Annex-III seems suitable for Para A.1.1, etc.			X	Annex-III is about reactor description and is already referred in Appendix Chapter 5 para A.5.3.
Annex-IV								
No comment								