*DS 511 Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors (Revision of SSG-22)*

| **COMMENTS BY REVIEWER**<br>**Reviewer:** **Page.**<br>**Country/Organization:** **Date: 01 October 2021** | | | | | **RESOLUTION** | | | |
|---|---|---|---|---|---|---|---|---|
| **Comment No.** | **Country Reviewer Comment No.** | **Para/Line No.** | **Proposed new text** | **Reason** | **Accepted** | **Accepted, but modified as follows** | **Rejected** | **Reason for modification/ rejection** |
| 1. | France 1 | Contents | part 3.14 – 3.17 line : page 14 or 15 not 145;<br>« references » line : page 80 not 800<br>« contributors to review » part : page 83 not 833 | pagination error | X | | | |
| 2. | Korea 1 (KINS) | Contents | The Regulatory ------ Research Reactors (3.2-3.6) ------ 11~~2~~<br>The Use of ---(3.14-3.17)--- 14~~5~~<br>References ------ 80~~0~~<br>Contributors to ------ 83~~3~~ | Correction of page number | X | | | |
| 3. | Germany 1- EPReSC (BMU/GRS) | General | **Postulated initiating events**<br>6.22. Requirements for identifying postulated initiating events for research reactors are established in Requirement 18 of SSR-3 [1]. | The headline „Postulated initiating events" on page 28 shall not be separated from the following text. It might be more convenient to place the headline on the following page.<br>See also headlines on pages | X | | | Will be resolved at editorial stage. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | … | 32, 35 and 64. | | | | |
| 4. | Germany 1-NUSSC | General | Specification of a research reactor with a low potential hazard should be consistent within the entire document. Please use consistently the complete formulation like "… For a research reactor, critical or subcritical assembly with a low potential hazard ~~such as a critical or subcritical assembly~~ … " | Referring only to critical or subcritical assemblies may be misleading | X | | | |
| 5. | Germany 2-NUSSC | General | As formulated in para 1.9 in scope of this Guide, it is about "the use of graded approach **in the application** of the safety requirements", to be in line with SSR-3. Please use this formulation all over the text, further variations of wording confuse and are misleading | Please use this formulation all over the text, further variations of wording, as for example "this requirement can be applied/cannot be applied using a graded approach" confuse and are misleading. | | X<br>The following text formulation is used "the way that this requirement is applied is the same irrespective of the potential hazard of the facility<br>Or<br>"the way this requirement is applied cannot be graded" | | For consistency with SSR-3. |
| **Section 1** | | | | | | | | |
| 6. | Germany 3-NUSSC | 1.6 | The Safety Guide provides recommendations on the use of a graded approach in the application of the safety | We like to emphasize the importance that the objective of grading is to balance the stringency of regulatory | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | requirements established in SSR-3 [1] for research reactors, including critical assemblies and subcritical assemblies, without compromising safety. | requirements with the associated risk without compromising safety. | | | | |
| 7. | Germany 4-NUSSC | 1.10 | This Safety Guide is primarily intended for use for heterogeneous, thermal spectrum research reactors having a power rating of up to several tens of megawatts. For research reactors of higher power, 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), additional guidance may be needed. Homogeneous reactors and accelerator driven systems are out of the scope of this publication. | Compare with para 1.8 of SSR-3. The scope of the Safety Guide should be the same as of the Safety Requirement. | X | | | |
| 8. | Germany 1-WASSC (BMU/GRS-BASE) | 1.9 | This Safety Guide considers the application of a graded approach throughout the lifetime of a research reactor without decommissioning (site evaluation, design, | According to the Glossary and GSR Part 6 decommissioning is part of the lifetime. “the terms siting, design, construction, commissioning, operation and decommissioning are normally | | | X | The scope is in line with the approved DPP. Such detail is not needed here. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | construction, commissioning, operation and preparation for decommissioning), including utilization and experiments that are specific features of research reactor operation. | used to delineate the six major stages of the lifetime of an authorized facility." | | | | |
| **Section 2** | | | | | | | | |
| 9. | Germany 5-NUSSC | 2.7 | The overall method to determine the graded approach may be qualitative, quantitative, ~~qualitative~~ or a combination of both. The graded approach presented in this Safety Guide has two steps. First is the qualitative categorization of the facility in accordance with its potential hazard (see para. 2.16 of SSR-3 [1]). Second is consideration of a specific safety requirement from SSR-3 [1], and the quantitative and/or qualitative analysis of any activities and/or SSCs associated with that requirement. The use of a graded approach by the operating organization shall | Please put "qualitative" first, as this is the order later in text<br><br>As in para 6.28 of SSR-4 and in line with para 3.15 of SF-1 | | X<br>The overall method to determine the graded approach may be qualitative, quantitative, ~~qualitative~~ or a combination of both…… and the quantitative and/or qualitative analysis of any activities and/or SSCs associated with that requirement. | | The additional text is covered in para 2.1 and 2.5 among others of this safety guide and is in line with the SSR-3 para 6.18. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | be justified in accordance with the categorization of the facility, which shall be subject to review by the regulatory body. | | | | | |
| 10. | Germany 6-NUSSC | 2.9 Line 20 | …<br>On the basis of these characteristics, together with the application of expert judgement and consideration of any other factors that might affect the potential ~~radiological~~ hazard, the research reactor should be categorized as a high, medium or low potential hazard. | Clarification in order to bring in line with para. 2.16 of SSR-3 and para 2.7 of current document | X | | | |
| 11. | Germany 7-NUSSC | 2.9 A New issue | A useful tool for the categorization of the facility in accordance with its potential hazard is an assignment of a research reactor to a cooling category as following:<br>(a) After shutdown from full power operation the reliability of active cooling systems must be ensured to remove the residual heat from the reactor core to an ultimate heat sink. In the worst-case scenario cladding failure and melting of fuel element | From our practical experience we know that determining the radiological hazard potential is the most demanding and crucial task in applying a graded approach.<br>We suggest to add an assignment of a research reactor to a cooling category as a useful practical tool for the categorization of the facility.<br>We believe that Member States would benefit from more guidance on this topic. | | | X | Graded approach in accordance with potential hazards is described. Among others cooling is one factor to consider in deciding application of graded approach to certain requirements. Similar description to apply graded approach to cooling system is |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | shall be considered.<br><br>(b) After shut-down from full power operation the reliability of passive cooling systems must be ensured to remove the residual heat from the reactor core to an ultimate heat sink. In the worst-case scenario cladding failure and melting of fuel element shall be considered.<br><br>(c) After shut-down from full power operation no cooling systems are necessary for residual heat removal from the reactor core to an ultimate heat sink. In the worst-case scenario, no cladding failure or melting of fuel element occurs. | | | | | covered in para 6.3 (b) of this Safety Guide. |
| 12. | Germany 8-NUSSC | Introduce a new para (2.10 A) | 2.10. Following the categorization of the facility in step 1, an analysis should be performed to determine the appropriate manner for meeting a specific safety requirement using a graded approach. A safety requirement may address a | Please introduce a new para. at the beginning of Step 2: Analysis and Application of a Graded Approach.<br>This include a very useful information, which is missing in Chapter 2. Compare also with para. 2.8. of SSG-22. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | specific SSC, or an element of the management system. The safety significance of each SSC or management system element (including SSCs and management system elements related to experiments) can be determined through the step 2 analysis. Requirement 16 of SSR-3 [1] states that "All items important to safety for a research reactor facility shall be identified and shall be classified on the basis of their safety function and their safety significance".<br>2.10 A. In this step, the level of detail at which requirements are applied to activities and/or SSCs is determined, in accordance with the importance to safety of the activity or SSC. The level of detail should cover, for example, the rigour of the analysis to be conducted, the frequency of activities such as testing and preventive maintenance, the stringency of required approvals and the degree of oversight of activities. | | | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 13. | Germany 9-NUSSC | 2.11 | The ~~safety function and~~ safety significance ~~and potential risks~~ of SSCs should be determined by conducting a safety assessment (see DS510A [10]) by analyzing the consequences of a failure of the intended safety function to be performed by the considered SSCs. ~~When identifying SSCs that are important to safety, classifying them by their importance to safety, and then considering a graded approach in their design, para 6.32 of SSR-3 [1] states that "The basis for the safety classification of the structures, systems and components shall be stated and the design requirements shall be applied in accordance with their safety classification." The application of design requirements commensurate with the safety classification of an SSC is the basis of a graded approach in the design process.~~ Based on the safety class appropriate design requirements should | The application of an appropriate methodology for safety classification (e.g. following the proposed methodology of SSG-30) will directly lead to an appropriate safety class commensurate with the safety significance of the SSCs. The safety significance is based on the consequences in case of a failure of the intended safety functions and additional factors (such as frequency, time before countermeasures are due, etc.) taken into account. This process includes implicitly a graded approach and an additional application of the graded approach on the safety classification is not necessary. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | be assigned to meet para 6.32 of SSR 3 [1]. | | | | | |
| 14. | Germany 10-NUSSC | 2.14 | Specific recommendations on the use of a graded approach in the application of each safety requirement of SSR-3 [1] are provided in Sections 3–8, including ~~on~~ requirements to which a graded approach cannot be applied. Examples are given for the graded application of requirements for research reactors with a high, medium, or low potential hazard. | Clarification | X | | | |
| 15. | France 2 | 2.12 - (b) | the safety, health, environmental, security, quality, human-and-organizational-factor, societal and economic objectives of the operating organization | Reference to requirement 4 of SSR-3 which contains human-and-organizational-factor, societal | | | X | GSR Part 2 requirement para is qouted. |
| 16. | France 3 | 2.9 - (j) | “The site evaluation, including external hazards (natural, man-made and hazard combinations) associated with the site and the vicinity of the research reactor including proximity to population groups”. | A list of external hazards would be beneficial to avoid oversights (in particular man-made hazards and combinations).<br><br>The vicinity of the RR implies larger scope for categorizing the RR, proximity to population groups is not sufficient, environmental | | | X | Para from SSR-3 is quoted. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | | stakes and industrial environment could impact classification | | | | |
| **Section 3** | | | | | | | | |
| 17. | Germany 11-NUSSC | 3.1 | General requirements for the legal and regulatory infrastructure for facilities and activities are established in IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [16], which includes ~~ing~~ requirements | Wording | X | | | |
| 18. | Korea 2 (KINS) | 3.6/6-8 | in IAEA Safety Standards Series No. GSG-12~~, Organization, Management and Staffing of the Regulatory Body for Safety~~ [19] and GSG-13 [17], respectively. | For consistency | | | X | IAEA style of writing, GSG-12 is referred first time. |
| 19. | Germany 12-NUSSC | 3.8 (c) | Issue of permits and licences, for the various stages; | Further specification is needed. Compare also with para. 3.8. of SSG-22. | X | | | |
| 20. | Germany 13-NUSSC | 3.9 | The steps in the authorization process apply to all research reactors at all stages of their lifetime and ~~may~~ should apply to experiments and | Clarification | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | modifications depending on their importance to safety | | | | | |
| 21. | Germany 14-NUSSC | 3.13 | The ~~use of~~ complementary probabilistic safety assessment~~, as appropriate~~, which might be carried out to supplement deterministic safety analysis if appropriate (see Requirement 5 of SSR-3 [1]), is another element of the safety analysis report requirement that could vary in accordance with the potential hazard of the facility. | According to SSR-3 probabilistic safety assessment (PSA) are complementary to deterministic safety analyses. The PSA shall be performed where appropriate. | X | | | |
| **Section 4** | | | | | | | | |
| 22. | Germany 15-NUSSC | 4.2 Line 18 | … There are elements of this requirement to which graded approach cannot be applied ~~using a graded approach~~, for example, for the operating organization to have prime responsibility for the safety of the research reactor, and the requirement to develop and sustain a strong culture for safety. | Please put wording in line with SSR-3, the same for further paras.<br>Please see General comment as well, for consistency within the document. | | X | | Please see resolution to Germany comment 2 (NUSSC). |
| 23. | Germany 16-NUSSC | 4.3 Line 5 | … In a facility with a low potential hazard~~, such as some subcritical~~ | The term low hazard potential is more comprehensive and does not stipulate the idea that | | X<br>….In a facility such | | For completeness. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~assemblies,~~ the requirement for sufficient staff could result in a small operating organization, with the necessary training to operate, maintain, and ensure the safety of the research reactor. | critical and subcritical assemblies are per se of a low hazard potential (which they are not). Please delete this example. | | as some low potential hazard research reactors, critical and subcritical assemblies…….. | | |
| 24. | Pakistan 1-NSGC (PAEC) | 4.5 | The requirement to establish and implement a safety policy cannot be applied using a graded approach. The safety policy is a central component of the management system for any **nuclear facility**, to ensure that all activities within the operating organization give safety the highest priority. | Safety policy is generic document for all nuclear facilities including NPP/RR and other nuclear facilities, it is not made specific to RRs. So it is proposed to use term ‘nuclear facilities’ rather research reactors. | X | | | |
| 25. | Germany 17-NUSSC | 4.6 | Requirements for the management system for a research reactor facility are established in Requirement 4 of SSR-3 [1]. ~~Paragraph 4.7 of SSR-3 [1] states that “The level of detail of the management system that is required for a particular research reactor or experiment shall be governed by the potential hazard of the reactor and~~ | Further specification of the requirement is needed. Compare also with para. 4.5. of SSG-22. | | X Requirements for the management system for a research reactor facility are established in Requirement 4 of SSR-3 [1]. ~~Paragraph 4.7 of SSR-3 [1] states that “The level of detail of the management system that is required for a particular research~~ | | Requirement 7 and para 4.15 of GSR Part 2 is referred. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~the experiment".~~ According to the para. 4.7. of SSR-3 [1] the complexity of the management system for a particular research reactor and associated experimental facilities should be commensurate with the potential hazard of the reactor and the experimental facilities, and the requirements of the regulatory body. Requirement for the preparation and implementation of a graded management system is established in Requirement 7 of GSR Part 2 [14], which state that grading of the application of management system requirements is required to be applied to the products and activities of each process and that the grading is required to be such as to deploy appropriate resources, on the basis of consideration of:<br>—The safety significance and complexity of each activity;<br>—The hazards and the | | | ~~reactor or experiment shall be governed by the potential hazard of the reactor and the experiment".~~ According to the para. 4.7. of SSR-3 [1] the complexity of the management system for a particular research reactor and associated experimental facilities should be commensurate with the potential hazard of the reactor and the experimental facilities, and the requirements of the regulatory body. Requirement for the preparation and implementation of a graded management system is established in Requirement 7 and para 4.15 of GSR Part 2 [14]. | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | magnitude of the potential impact (risks) associated with the safety, health, environmental, security, quality and economic elements of each activity;<br>—The possible consequences if an activity is carried out incorrectly. | | | | | |
| 26. | Pakistan 2-NSGC (PAEC) | 4.8 | A procedure for a simple maintenance task on a component in a non-active system with low safety significance could be written by **an experienced person** and reviewed by maintenance supervisor. | The developing procedure by experienced member of the engineering personnel will limit the opportunities for maintainers. It is therefore proposed that it should remain generic. | | X<br>A procedure for a simple maintenance task on a component in a non-active system with low safety significance could be written by an experienced member of the ~~engineering~~ operating personnel and reviewed by a maintenance supervisor | | Using terminology in compliance with SSR-3 para 7.59. |
| 27. | Germany 18-NUSSC | 4.11 Line 10 | … For a research reactor with a low potential hazard, the management system could consist of relatively few processes and procedures, and an audit of the management system could occur as part of the renewal of the | Some countries grant indefinite operational licenses, here the audit of the management system could be coupled to the periodic safety review | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | authorization from the regulatory body or the periodic safety review. | | | | | |
| 28. | Pakistan 3-NSGC (PAEC) | 4.12 | These requirements can be applied using a graded approach, for example, by taking the potential hazard of the research reactor into account when determining the frequency and scope of safety assessments (such as self-assessments, **independent assessment** and peer reviews) throughout the lifetime of the facility. | In post Fukushima scenario, corporate independent assessment of nuclear facilities is becoming industry practice. Therefore it may be including as part of self-assessment or separate. This is in addition to facilities own self-assessment. | X | | | |
| 29. | Germany 19-NUSSC | 4.14 Line 6 | … A minimum list of items that the reactor safety committee is required to review is provided in para 4.27 of SSR-3 [1] (see also paras 7.8 and 7.9 of this Safety Guide). | Clarification | X | | | Relevant paras are referred (7.9 and 7.10). |
| **Section 5** | | | | | | | | |
| 30. | Germany 20-NUSSC | 5.3 New issue | Paragraphs 4.1–4.5 of SSR-1 [15] develop the basis for applying a graded approach to the various site related evaluations and decisions, commensurate with the radiological hazard of the research reactor. The main | Please add this important issue | | | X | The existing text 5.3 (g) already cover this issue. It is quotation from safety requirement and cannot be changed. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | factors to be considered in site evaluation are the following:<br>….<br>(g) The potential for on-site and off-site consequences in the event of an accident. <u>In addition, the dispersion in air and water of radioactive material released from the nuclear installation in operational states and in accident conditions shall be assessed according to Requirement 25 of SSR-1.</u> | | | | | |
| 31. | Japan 1-NUSSC (NRA) | 5.7. | For the evaluation of hazards associated with human induced events in site evaluation for a research reactor, only <u>one intensity level</u> for each event is expected to be considered in the design basis. Recommendations on the screening and analysis of hazards associated with human induced events are provided in IAEA Safety Standards Series No. DS520, Hazards Associated with Human induced External Events in Site Evaluation for Nuclear Installations [25]. | Clarification.<br>"One intensity level" is unclear, as the preceeding wordings in the previous draft (step 8; see the below) was deleted in the step 11 draft. Furthermore, intensity level is not addressed in DS520.<br><br>*"5.8. Human induced events cannot be included in site evaluation using the same approach as other external events. <u>Because human induced events are discrete and are not characterised by a range of frequency and severity, only one intensity level</u> for each event is expected for consideration in the design* | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | | *basis. Recommendations on site survey and site selection …,"* | | | | |
| **Section 6** | | | | | | | | |
| 32. | Germany 21-NUSSC | 6.2 Line 7 | The use of a graded approach should result in design features that fully meet this requirement and are appropriate for the potential hazard from the research reactor. Graded approach cannot be applied to two elements of this requirement, which are shielding against radiation and control of planned radioactive discharges during normal operation. The design of shielding for protection from radiation should be based on the radiation protection limit values, which are not subject of graded approach. The control of radioactive discharges (see Requirements 59 and 64 of SSR-3 [1]) is necessary to protect the public and the environment and to meet regulatory requirements, and this requirement ~~cannot be applied using a graded~~ | The design of shielding for protection from radiation should be based on the dose limits / dose constraints and on magnitude of the radiation hazard. The necessary shielding is the result of the design process and not an application of the graded approach. | | | X | Justification is technically correct and addressed in existing text para 6.3 c (ii). |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~approach~~ is not subject of graded approach as well. | | | | | |
| 33. | Germany 22-NUSSC | 6.3 (a) New issue | (ii) Some research reactors may have inherent self-limiting power levels and/or systems that physically limit the amount of positive reactivity that can be inserted into the core. This property can be used for graded approach in the design of the shutdown system. | Please introduce additional bullet. This include a very useful information, which is missing in para. 6.3. (a). Compare also with para. 6.6. of SSG-22. | X | | | |
| 34. | Germany 23-NUSSC | 6.3 (b) (i) | For some research reactors (typically with a medium or high potential hazard and higher power) a forced convection cooling system to remove fission heat, could be necessary to meet the acceptance criteria for the design, in all operating conditions and accident conditions, whereas for research reactors with less demanding cooling needs~~, such as some critical and subcritical assemblies,~~ fission heat could be generated at sufficiently low levels that it could be adequately removed without the need for an engineered system. | The term “with less demanding cooling needs” is more com-prehensive and does not stipulate the idea that critical and subcritical assemblies are per se of a low hazard potential (which they are not) | | | X | It is preferred to leave it as it is for clarification the text ‘some’ is already used. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 35. | Germany 24-NUSSC | 6.3 (c) Step 11 | A graded approach can be used in the application of some elements of Requirement 7 of SSR-3 [1] for the main safety functions, as follows:<br><br>(c) Confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases:<br>….<br>(ii) <u>Graded approach cannot be applied to the requirement of shielding.</u> The design of shielding against radiation should be based <u>on the dose limits / dose constraints and</u> on the magnitude of the radiation hazard calculated for each location in the research reactor where actions by operating personnel are necessary in operational states and in accident conditions, and for appropriate locations outside the research reactor. The appropriate material and thickness of shielding that is commensurate with | The design of shielding for protection from radiation should be based on the dose limits / dose constraints and on magnitude of the radiation hazard. The necessary shielding is the result of the design process and not an application of the graded approach. | | | X | Please see resolution to Germany comment 21. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | the hazard can then be included in the design. | | | | | |
| 36. | Pakistan 4-NSGC (PAEC) | 6.3 (b) (iii) | The scope and necessity of coolant systems (see Requirement 47 of SSR-3 [1]), including emergency core cooling systems to **make up** the inventory of reactor coolant in the event of a loss of coolant accident ……………………… | ‘make up’ is the appropriate and relevant term instead of ‘replace’, so it is proposed to be used. | X | | | |
| 37. | Germany 25-NUSSC | 6.4 Line 5 | … Specific design provisions, or SSCs included in the design to protect reactor personnel and the public from radiation (e.g. an emergency filtration system) could be larger and/or more complex for a research reactor with a higher potential hazard. | Clarification | X | | | |
| 38. | Germany 26-NUSSC | 6.6 Line 4 | … The quantity of information that would be adequate to decommission a research reactor with a higher potential hazard should be larger in scope than for research reactors with a lower potential hazard ~~(e.g. some low power reactors, critical assemblies, subcritical assemblies)~~. | Clarification. See also general comment. | | | X | Please see resolution to Germany comment 23. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 39. | Germany 27-NUSSC | 6.8 - 6.9 | 6.8. Defence in depth is an important design principle that is required for all research reactors regardless of potential hazard. ~~; However, this requirement should be applied using a graded approach by recognizing that for low power research reactors, or critical and subcritical assemblies, accidents which need mitigation by the fourth or fifth level of defence in depth (see para. 2.12 of SSR-3 [1]) may not be physically possible.~~<br>~~6.9~~. For a facility with a low or medium potential hazard, the first four levels of defence in depth should be included in the design. The design capability of the engineered safety features can use a graded approach, for example the decay heat load could be smaller, and typically a smaller fission product inventory needs to be confined or mitigated than for a research reactor with a high potential hazard. | Defence in depth is important and shall be applied regardless of the hazard potential of a research reactor. To clarify this issue, we proposed to combine 6.8 and 6.9. | | X<br>Defence in depth is an important design principle that is required for all research reactors regardless of the potential hazard. ~~however, it should be recognized that for low power research reactors, critical assemblies and subcritical assemblies, the types of accident that the fourth or fifth level of defence in depth are intended to cope with might not be physically possible~~.<br>~~For a facility with a low or medium potential hazard,~~ The first four levels of defence in depth should be included in the design. The design capability of | | For clarity and completness. |

| | | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | | | | | | | the engineered safety features can use a graded approach, for example the decay heat load could be smaller, and typically a smaller fission product inventory needs to be confined or mitigated than for a research reactor with a high potential hazard.<br><br>~~however,~~ It should be recognized that for low power research reactors, critical assemblies and subcritical assemblies, the types of accident that the fourth or fifth level of defence in depth are intended to cope with might not be physically possible. | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 40. | Germany 28-NUSSC | 6.9 | ~~For a facility with a low or medium potential hazard, the first four levels of defence in depth should be included in the design. The design capability of the engineered safety features can use a graded approach, for example the decay heat load could be smaller, and typically a smaller fission product inventory needs to be confined or mitigated than for a research reactor with a high potential hazard.~~ | Please remove this to para 6.8.<br><br>SSR-3 requires consideration of design extension condition for all research reactors. The requirement cannot be waved, only the way how it is fulfilled should be graded.<br>Even if accidents with a core damage may practically be excluded, mitigation of any unnecessary radiological exposure to people and environment should be applied. For this reason, a general statement that level five of defence in depth is not applicable for research reactors with lower potential hazard may be misinterpreted. In addition, to our understanding level 5 of DiD is not a further escalation of level 4 of DiD, but may also be necessary, depending on the research reactor, starting from level 3 of DiD. This means, level 5 of DiD has to be seen more parallel to levels 3 and 4 of DiD. | | X | | Please see resolution to Germany comment 27. |
| 41. | Germany 29-NUSSC | 6.16<br>Line 15 | … For a research reactor with a low potential hazard, ~~such as a subcritical assembly~~, there might not be a significant hazard from | Clarification.<br>See also general comment. | | X<br>For a research reactor with a low potential hazard, such as <u>some critical and</u> subcritical | | Please see resolution to Germany comment 23. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | activation products. | | | assemblies, there might not be a significant hazard from activation products. | | |
| 42. | Germany 30-NUSSC | 6.21 Line 2 | … Although it is not possible to use graded approach to ~~apply~~ this requirement ~~using a graded approach~~, the design basis for items important to safety in a facility with a low potential hazard ~~research reactor, critical assembly or subcritical assembly with a low potential hazard,~~ is typically less complex, and requires less analysis to demonstrate that its performance meets acceptance criteria, than in a facility with a high potential hazard. | Clarification | X | | | |
| 43. | Japan 2-NUSSC (NRA) | 6.23. /4 | A comprehensive set of postulated initiating events is always required for the safety analysis of a research reactor regardless of potential hazard, and are required to be identified on the basis of engineering judgement, ~~operating experience feedback~~. operating experience feedback (including | Wording/Editorial. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | operating ~~operational~~ experience from similar facilities) and deterministic assessment, complemented, where appropriate and available, by probabilistic methods: see para. 6.36 of SSR-3 [1]. | | | | | |
| 44. | Germany 2-WASSC (BMU/GRS-BASE) | 6.23 | A comprehensive set of postulated initiating events is always required for the safety analysis of a research reactor regardless of potential hazard, and are required to be identified on the basis of engineering judgement, operating experience feedback~~. operating experience feedback~~ (including operational experience from similar facilities) and deterministic assessment, complemented, where appropriate and available, by probabilistic methods: | Wording | X | | | |
| 45. | Pakistan 5-NSGC (PAEC) | 6.31-6.32 | Addition of para regarding procedures / guidelines for Design Extension Conditions<br>It is proposed that paragraph(s) addressing guidelines / procedures for handling design extension | EOP are used in design basis accidents to control accident and mitigate consequence. However, in case these are not controlled and design extension conditions approaches, the additional text / para may be drafted / | | | X | Not the scope of this Safety Guide.<br><br>It is addressed in other Safety Guide i.e. SSG-20. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | conditions be added in the section / document to address the guidance for handling design extension conditions. | included aiming to provide the required guidelines / procedures to deal with the design extension conditions. | | | | |
| 46. | Germany 31-NUSSC | 6.32 Line 4 | … In a research reactor with a low potential hazard ~~such as a subcritical assembly~~ with few SSCs important to safety, accidental criticality could be the only event included in the analysis of design extension conditions. | Clarification. See also general comment. | | X … In a research reactor with a low potential hazard such as some critical and subcritical assemblies with few SSCs important to safety, accidental criticality could be the only event included in the analysis of design extension conditions. | | Please see resolution to Germany comment 23. |
| 47. | Germany 32-NUSSC | 6.35 Line 5 | … For a research reactor with a low potential hazard ~~such as a critical assembly~~ where the irradiated fuel can be safely stored in air, the safety analysis may demonstrate that no engineered safety feature is necessary to maintain fuel integrity in response to a loss of coolant accident. | Clarification. See also general comment. | | X … For a research reactor with a low potential hazard such as some critical assemblies where the irradiated fuel can be safely stored in air, the safety analysis may demonstrate that no engineered safety feature is necessary to maintain fuel integrity in response to a loss of coolant accident. | | Added ‘some’ as in resolution to Germany comment 23. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 48. | Korea 3 (KINS) | 6.39/9 | Recommendations on the application ……in paras 6.40-~~6.48~~ 6.47 | Paras. 6.40-6.47 correspond to the principles (a)-(e) in the para. 6.39 | | X | | Relevant paras are referred. |
| 49. | Germany 33-NUSSC | 6.41 | Graded approach cannot be applied to the ~~This~~ requirement that no single failure should prevent~~s~~ SSCs in a safety group from performing a main safety function~~, cannot be applied using a graded approach.~~ For all research reactors, the groups of equipment delivering any one of the main safety functions are required to be designed with appropriate redundancy, independence and diversity to ensure high reliability. However, the required degree of redundancy can be graded and may be lower for a low hazard potential. | The degree of redundancy may be lower for a low potential hazard facility than for a high potential hazard facility. | | X<br>Graded approach cannot be applied to the ~~This~~ requirement that no single failure should prevent~~s~~ SSCs in a safety group from performing a main safety function~~, cannot be applied using a graded approach.~~ For all research reactors, the groups of equipment delivering any one of the main safety functions are required to be designed with appropriate redundancy, independence and diversity to ensure high reliability. However, the required degree of redundancy can be graded and may be lower for a low potential hazard facility. | | For completeness. |
| 50. | Germany 3-WASSC (BMU/GR | 6.55 | For a research reactor with a high potential hazard and a large number of | The word complex doesn't sound very well with the design of escape routes. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | S-BASE) | | operating personnel, the design of escape routes could be relatively ~~complex~~ versatile and the location where personnel assemble could need specific design features to protect personnel from hazards during an emergency. | | | | | |
| 51. | Korea 4 (KINS) | 6.58/5 | (a)----The need for disposal facilities ……. will ~~also be~~ likely ~~to~~ be minimal. | Reflect the previous resolution table for No. 130 | X | | | |
| 52. | Germany 34-NUSSC | 6.61 | Paragraph 6.94 of SSR-3 [1] requires that adequate provision is made for shielding, ventilation, filtration and decay systems in the design of a research reactor. The design of ventilation systems can use a graded approach based on the potential radiological hazard and the necessary occupancy of the room in operational states and in accident conditions. For a research reactor with a low or medium potential hazard, the number of locations within the facility requiring ventilation systems to mitigate radiological hazards is typically fewer than in a research reactor with a high | The design of shielding for protection from radiation should be based on the dose limits / dose constraints and on magnitude of the radiation hazard. The necessary shielding is the result of the design process and not an application of the graded approach. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | potential hazard. ~~Similarly, the design calculations and features necessary to ensure adequate shielding of SSCs with high radiation fields, should be fewer and less complex.~~ | | | | | |
| 53. | Germany 35-NUSSC | 6.62 | Design provisions to monitor and control access to SSCs ~~with~~ imposing radiological hazards to workers can be applied using a graded approach. | Clarification | X | | | |
| 54. | Germany 36-NUSSC | 6.65 Line 4 | … In all cases, the analysis of the human– machine interface should consider all normal operational states, postulated initiating events, design basis accidents and selected, but enveloping design extension conditions, to ensure that combinations of alarms and indications in the control room are unambiguous. | Addition in order to make sure that all alarms and signal are covered. | X | | | |
| 55. | Germany 37-NUSSC | 6.67 | (b) New utilization and modification projects, including experiments that have a significant effect on ~~major significance for~~ safety…. | Clarification | | X “…including experiments that have a major or significant effect on safety……” | | For consistency with SSG-24. |
| 56. | Germany 38-NUSSC | 6.68 Line 7 | … The analysis of the modification should be | Clarification | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | and<br>Line 12 | reviewed by the reactor safety committee and the regulatory body…<br>… This analysis should be reviewed by the reactor safety committee and approved by the reactor manager before the design process proceeds. | The same for para. 7.10. | | | | |
| 57. | Germany<br>39-NUSSC | 6.75 | Research reactor designs normally include provisions to ensure safety during shutdown and typically these provisions can be used during a ~~extended~~ long shutdown. For all SSCs that are important to safety, and which could suffer degradation during the ~~extended~~ long shutdown period, provision should be made for a preservation programme that includes inspecting, testing, maintaining, dismounting and/or disassembling SSCs, as appropriate, during the shutdown period. As an alternative to implementing a preservation programme for installed equipment, it may be more practical to remove equipment; this decision is usually linked to the future of the research | This para is dealing with a long shutdown.<br>Extended shutdown is approached in paras 7.90 – 7.92 of current Guide. | X | | | For consistency text in para 7.90 is revised accordingly. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | reactor. All modifications made to a research reactor in ~~extended~~ <u>long</u> shutdown are also subject to Requirements 36 and 83 of SSR-3 [1], including review, assessment and approval by the regulatory body prior to implementation, when appropriate. | | | | | |
| 58. | Germany 40-NUSSC | 6.76 Line 10 | … For a research reactor with a low potential hazard, ~~such as a subcritical assembly~~ with irradiated fuel containing a low fission product inventory that does not need shielding or water cooling, | Clarification.<br>See also general comment. | | X<br>… For a research reactor with a low potential hazard, such as <u>some critical and</u> subcritical assemblies with irradiated fuel containing a low fission product inventory that does not need shielding or water cooling, | | Please see resolution to Germany comment 23. |
| 59. | Germany 41-NUSSC | 6.84 | (c) The use of conservative methods and criteria is a mean~~s~~ of simplifying the safety analysis. | Wording<br>“is a mean of” | X | | | |
| 60. | Germany 42-NUSSC | 6.87 | A graded approach can <u>not</u> be used for the design of shielding throughout the research reactor, ~~based on the number of rooms where SSCs could be a source of radiation in operational states or in accident~~ | The design of shielding for protection from radiation should be based on the dose limits / dose constraints and on magnitude of the radiation hazard. The necessary shielding is the result of the design process and not an | | | X | The graded approach is applicable to requirement, the examples in the text are kept for explanation. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~conditions, and on the characteristics of the radiation risk~~. In accordance with Requirement 42 of SSR-3 [1], the buildings and structures are required to be designed to maintain radiation levels as low as reasonably achievable. For a research reactor with a high potential hazard, a larger number of rooms where equipment associated with reactor operation, isotope production, experimental devices or radioactive waste storage could need to be provided with shielding as part of the building design. In a facility with a lower potential hazard, with a small number of rooms where a radiation risk is present, the design of structures to provide adequate shielding could be less complex. | application of the graded approach. | | | | |
| 61. | Germany 43-NUSSC | 6.93 | For a research reactor with a high potential hazard, monitoring of parameters such as temperature, flow and radiation levels in each fuel channel, could be | Design feature, in singular, is more suitable, as monitoring of parameters is a design feature here. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | design feature~~s~~ that ensure an automatic response from the reactor protection system, or an action by operating personnel in response to an alarm. Such design feature~~s~~ could be necessary…. | | | | | |
| 62. | Germany 44-NUSSC | 6.103 | The requirement to monitor and control the properties of the reactor coolant (e.g. the pH and conductivity: see para. 6.162 of SSR-3 [1]) is applicable to all water-cooled research reactors of any power level ~~including subcritical assemblies,~~ to ensure that water conditions do not degrade reactor SSCs important to safety, especially boundaries that prevent the release of fission products, such as the fuel cladding. | Already covered by para. 1.1 | | X<br>The requirement to monitor and control the properties of the reactor coolant (e.g. the pH and conductivity: see para. 6.162 of SSR-3 [1]) is applicable to all water-cooled research reactors of any power level including <u>some</u> subcritical assemblies, to ensure that water conditions do not degrade reactor SSCs important to safety, especially boundaries that prevent the release of fission products, such as the fuel cladding. | | Please see resolution to Germany comment 23. |
| 63. | Iran 1-EPReSC (INRA) | Paragraph 6.105/ First line | "The need for an emergency core cooling system should be defined in the design stage…" | Is there any reason for omitting "core"? | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 64. | Germany 45-NUSSC | 6.105 Line 8 | … For a facility with a low potential hazard, ~~such as some subcritical assemblies, where the irradiated fuel is normally stored in dry conditions,~~ safety analysis could demonstrate that no emergency core cooling system is necessary to mitigate the consequences of a loss of coolant accident. | Deletion to avoid a predetermination on a specific design.<br>It is not clear why the conditions of the irradiated fuel influence the emergency core cooling needs. | | … For a facility with a low potential hazard, such as some subcritical assemblies, ~~where the irradiated fuel is normally stored in dry conditions,~~ safety analysis could demonstrate that no emergency core cooling system is necessary to mitigate the consequences of a loss of coolant accident. | | Examples retained as 'some' is used. Please see resolution to Germany comment 23. |
| 65. | Germany 46-NUSSC | 6.109 Line 5 | … This measurement is typically not necessary in a <u>research reactor that does not need an active water cooling</u>. ~~critical assembly or a subcritical assembly~~ | Here, the important aspect is the forced cooling of the core. Many water cooled research reactors do not require an active water cooling and hence do not require to measure the pressure across the core.<br>Deletion to avoid a predetermination on a specific design. | X | | | |
| 66. | Germany 47-NUSSC | 6.111 Line 5 | … For research reactors that operate for only a few hours per week or less frequently~~, such as some critical assemblies, a lower level, i.e.~~ two channel (one-out-of-two), redundancy can be applied, thus reducing the complexity of the design and of operation~~,~~ | 1) Statement is unclear. There are also smaller research reactors that operate only few hours a week. Giving solely critical assemblies as an example is misleading. See also general comment.<br><br>2) Costs must not be considered as a factor for | | X<br>… For research reactors that operate for only a few hours per week or less frequently, such as some critical assemblies, a lower level, i.e. two channel (one-out-of-two), | | For completeness the text deleted and examples are kept for additional clarification. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~as well as costs~~. | grading safety requirements. The safety of research reactor have to be assured due to the design and operation. | | redundancy can be applied, thus reducing the complexity of the design and of operation~~, as well as costs~~. | | |
| 67. | Germany 48-NUSSC | 6.113 | A graded approach can be applied to the reactor protection system, based on the potential hazard of the facility and the ~~number~~ kind of initiating events identified in the safety analysis (based on considerations of e.g. potential consequences of the hazard, time constrains, mitigating passive safety features). …. | The number of events is no relevant argument for a graded approach. | X | | | |
| 68. | Germany 49-NUSSC | 6.113 A New para | Regardless of the hazard potential of a research reactor, the reactor protection system should be designed in such a way that neither a single failure nor a common cause failure will prevent execution of mandatory safety functions. Consequently, graded approach cannot be applied to paras. 6.176, 6.177 and 6.181 of SSR-3 [1]. | Paras 6.176, 6.177 and 6.181 of SSR-3 includes important requirements related to the application of the single failure event, consideration of common cause failures or diversity for computer-based systems. To ensure the high reliability of the reactor protection system these three paras. are important and it should be clearly stated that a grading is not permitted. | | X<br>Regardless of the hazard potential of a research reactor, the reactor protection system should be designed in such a way that neither a single failure nor a common cause failure will prevent ~~execution of mandatory~~ meeting required safety functions. | | 'Mandatory' is not defined. The remaining text is covered by paras 6.40-6.41. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 69. | Pakistan 6-NSGC (PAEC) | 6.121 | Requirements for emergency response facilities on the site of a research reactor are established in Requirement 55 of SSR-3 [1]. **Accordingly, emergency response system should be established commensurate with the potential hazards due to internal and external events**. | Proposed new text will describe the general requirement of EPR system. | | | X | Already addressed in the text of same para. |
| 70. | Germany 50-NUSSC | 6.124 | For a research reactor with a high potential hazard, where forced cooling is needed to remove decay heat, the level of redundancy and the number of separate channels in the emergency power supply system should be based on the results of safety analysis, ~~including the frequency of abnormal occurrences and accident conditions for which emergency power is needed~~. | For a research reactor with a high potential hazard the design of the electrical power supply system should only be based on the results of safety analysis. In the safety analysis the frequency of abnormal occurrences and accident conditions is already suitably considered. | X | | | |
| **Section 7** | | | | | | | | |
| 71. | Japan 3-NUSSC (NRA) | 7.13. | A graded approach could also be applied to the education level and operating ~~operational~~ | Wording/Editorial | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | experience of trainees, the content and duration of initial and continuing training, training materials, the assessment of completed training, and to qualification, which can depend on the complexity of the research reactor design, as well as the potential hazard, planned utilization, and available infrastructure. | | | | | |
| 72. | Germany 51-NUSSC | 7.17. | Operational limits and conditions are based on the reactor design and on the information from the safety analysis report; consequently, a graded approach should ~~will have~~ been used in the application ….. | Clarification | X | | | |
| 73. | Germany 52-NUSSC | 7.21 Line 3 | … For example, in a low power reactor, the coolant outlet temperature could be selected as the parameter relating to the fuel temperature for which a safety system setting is defined, while in a higher power reactor, to prevent the safety limits from being approached, a complex system of variables should have defined safety system | Please add this important issue | | X ……In addition two safety parameters e.g. pressure and flow may also be needed for detection of some design basis incidents. | | For technical precision and consistency. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | settings, such as the coolant outlet temperature, the inlet temperature, the coolant flow rate, the differential pressure across the core and the primary pump discharge pressure, as well as parameters from experimental facilities. In addition, two different actuation criteria (e.g. pressure and flow rate) may also be required with regard to the detection of incidents | | | | | |
| 74. | Germany 53-NUSSC | 7.31 Line 4 | … For example, research reactors, critical assemblies or subcritical assemblies with a low potential hazard ~~of~~ ~~and subcritical facilities~~ typically have fewer personnel in the operating group and less or no expertise on power rise tests and operation at high power levels | Clarification. Here, the mentioning of critical and subcritical assemblies makes sense as power rise tests and high-power operation are given as examples. Addition of critical assemblies for completion. | X | | | |
| 75. | Germany 54-NUSSC | 7.32 | Stage C of commissioning (power ascension tests and power tests up to rated full power as defined in para 3.17 and paras 5.30–5.37 of DS509A [2]) is not necessary for critical and subcritical assemblies with a low potential hazard, and the scope, extent, and | Clarification.<br>Stage C can also be graded for critical and subcritical assemblies with a low potential hazard. | | X<br>Stage C of commissioning (power ascension tests and power tests up to rated full power as defined in para 3.17 and paras 5.30–5.37 of DS509A [2]) is not necessary for some critical and | | Also added ‘some’. Please see resolution to Germany comment 23. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | duration of Stage C is much less for low power research reactors (i.e., that are typically of low potential hazard) compared to those of higher power levels. | | | subcritical assemblies with a low potential hazard, and the scope, extent, and duration of Stage C is much less for low power research reactors (i.e., that are typically of low potential hazard) compared to those of higher power levels. | | |
| 76. | Germany 55-NUSSC | 7.34 | The principles applied in commissioning for the initial approach to criticality, reactivity device calibrations, neutron flux measurements, determination of core excess reactivity and shutdown margins, power raising tests and testing of the containment system or other means of confinement are similar for all research reactors regardless of potential hazard and hence cannot be subject to a graded approach…. | For clarity. | X | | | |
| 77. | Germany 56-NUSSC | 7.39 a)<br><br>New footnote | … (a) The procedure for regeneration of an ion exchange system for producing demineralized water for a storage tank will be of low safety significance[x] and will | Please add this important issue.<br>We suggest here as a footnote | X | | | Addressed in the text instead of footnote. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | involve mature and simple technology. Consequently, the operating procedure governing this application can be simplified.<br><br>[x] In some cases, the ion exchange resins can be dried. Radionuclides may be released during the drying process. There are limits to be observed for radioactive discharges with the air. Therefore, the safety significance is not to be regarded as low. | | | | | |
| 78. | Germany 57-NUSSC | 7.41 Line 5 | … In a research reactor with a high potential hazard, the supplementary control room ~~could~~ should include more monitoring and control equipment than a shutdown panel | Clarification | X | | | |
| 79. | Germany 58-NUSSC | 7.42 | Requirements for material conditions and housekeeping for research reactors are established in Requirement 76 of SSR-3 [1]. High standards of material conditions and housekeeping, including cleanliness, accessibility, adequate lighting, appropriate storage | Housekeeping and cleanliness are always important irrespectively of the hazard potential. It contributes to safety working conditions and is also important for occupational health and safety. In addition, even research reactors with a low hazard potential are operated in radiation-controlled areas | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | conditions, and identification and labelling of safety equipment are required regardless of the potential hazard of the research reactor. ~~A research reactor of a low potential hazard and fewer SSCs important to safety, should involve less effort to maintain a high standard of housekeeping and cleanliness compared to those facilities of medium and high potential hazards with a larger number of SSCs.~~ | requiring also organizational measures to avoid e.g. contamination and activation, for example by avoiding generation of unnecessary radioactive waste in such areas | | | | |
| 80. | Germany 59-NUSSC | 7.49 | ~~A balance should be sought between the improvement in the detection of faults that is gained from more frequent testing, against the risk that testing could be performed incorrectly and leave the SSC in a degraded state, the degradation of SSCs as a result of the testing activity, and the reduced availability of the SSC while testing is performed. This consideration also applies for periodic maintenance. The frequency of replacement of SSCs~~ | While this paragraph provides useful information on the establishment of testing intervals no information on grading is provided. It is recommended to move this paragraph to DS509B. Para 7.42 of current Guide contains already a link to this guide for more recommendations on maintenance, periodic testing and inspection. | | X<br>A balance should be sought between the improvement in the detection of faults that is gained from more frequent testing, against the risk that testing could be performed incorrectly and leave the SSC in a degraded state~~, the degradation of SSCs as a result of the testing activity, and the reduced availability of the SSC while testing is performed~~. This | | To make text consistent to apply graded approach. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~subject to ageing degradation (e.g. due to high radiation levels) can be based on the feedback of operating experience, including that from other reactors, and on the basis of the results of research and development.~~ | | | consideration also applies for periodic maintenance. The frequency of periodic maintenance may also depend on potential hazards for example replacement frequency of SSCs is subject to ageing degradation ~~(e.g~~. due to level of ~~high~~ radiation hazards ~~levels).~~ ~~can be based on the feedback of operating experience, including that from other reactors, and on the basis of the results of research and development.~~ | | |
| 81. | Germany 60-NUSSC | 7.50 | ~~The period for which an SSC is permitted to be out of service while reactor operation continues is usually stated in the operational limits and conditions for the research reactor and can be based on the availability requirement for the SSC from the safety analysis. For example, outage times of any duration might not be acceptable for automatic shutdown systems, while~~ | While this paragraph provides useful information on the duration of non-availabilities of SSC no information on grading is provided. It is recommended to move this paragraph to DS509B. Para 7.42 of current Guide contains already a link to this guide for more recommendations on maintenance, periodic testing and inspection. | | X<br>The period for which an SSC is permitted to be out of service while reactor operation continues is usually stated in the operational limits and conditions for the research reactor and can be based on the availability requirement for the SSC from the safety analysis. Additional | | Text retained for useful information. For further guidance referred DS509B. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~outage times of up to several days might be acceptable for other systems, with appropriate compensatory measures (e.g. for a purification system monitoring the primary coolant pH, the system could be unavailable for several days, provided that pH measurements are taken manually each shift). The allowed outage time should depend on the extent to which safety is impacted, or the ease of applying compensatory measures.~~ | | | information is provided in DS509B.~~For example, outage times of any duration might not be acceptable for automatic shutdown systems, while outage times of up to several days might be acceptable for other systems, with appropriate compensatory measures (e.g. for a purification system monitoring the primary coolant pH, the system could be unavailable for several days, provided that pH measurements are taken manually each shift). The allowed outage time should depend on the extent to which safety is impacted, or the ease of applying compensatory measures.~~ | | |
| 82. | Germany 61-NUSSC | 7.52 | ~~Some maintenance, periodic testing and inspection activities are highly specialized and~~ | While this paragraph provides useful information on specialized maintenance, periodic testing and | | | X | Graded approach is applicable to resources. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~involve complex and sophisticated techniques: these activities are more likely to be necessary in more complex research reactor designs. Such activities are often performed by contracted experts external to the operating organization for the research reactor. Such outsourcing should be carefully considered by the operating organization to ensure that external support is secured and that resources will be available throughout the operating lifetime of the research reactor. Recommendations on the use of external contractors for the performance of maintenance, periodic testing and inspection are provided in DS509B [3].~~ | inspections of SSC no information on grading is provided. It is recommended to move this paragraph to DS509B. Para 7.42 of current Guide contains already a link to this guide for more recommendations on maintenance, periodic testing and inspection. | | | | |
| 83. | Germany 62-NUSSC | 7.55 | The safety significance of ~~C~~changes to research reactor core management and fuel handling procedures should be determined ~~are modifications of major safety significance.~~ DS510B [11] provides | Please reformulate. Otherwise, text is contradictory. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | recommendations on a method for determining the safety significance of modifications to a research reactor and this method is applicable to core management and fuel handling. A graded approach to the analysis and verification of proposed changes to core management and fuel handling activities may be appropriate, on the basis of the safety significance of these changes (see also paras 7.70–7.73 of this Safety Guide). | | | | | |
| 84. | Germany 63-NUSSC | 7.58 Line 4 | …. For example, a fire affecting the instrumentation in the control room of a research reactor with a high potential hazard could be identified in the safety analysis as an event with a potential high consequence, needing to be mitigated by special means ~~the automatic action of an inert gas extinguishing system, combined with manual firefighting from~~ | As inert gas extinguishing system might have toxical impact, this example (automatic inert gas extinguishing system for control rooms) is not an optimal one as it threatens life of the shift personnel. | X | | | |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | ~~trained personnel~~. A fire in an administrative area, with a low safety consequence identified in the safety analysis, could be mitigated by the deployment of hand-held fire extinguishers and the actions of firefighting personnel. | | | | | |
| 85. | Iran 2-EPReSC (INRA) | Paragraph 7.66/ Bullet c | "(c) The identification and classification of the ~~hazard~~ emergency in order to declare the applicable emergency class." | "Classification of the hazard" is not clear. Does it mean "classification of the emergency" or does it mean "assess the hazard"? | | X<br>The identification of hazard and emergency classification ~~of the hazard~~. | | For clarity. |
| 86. | Iran 3-EPReSC (INRA) | Paragraph 7.66/Bullet F/First and second line | "~~The number and type of external organizations (e.g. police, fire fighting services, ambulance services and medical facilities)~~ The emergency services that ~~are part of to~~ should be involved in the emergency response…" | In GSR Part 7, there is another term for "external organizations" that are involved in emergency response and its definition is included in IAEA Safety Glossary too. It is suggested to replace "external organizations" with the term "emergency services" in this paragraph with the following definition:<br>"**emergency services**<br>**The local off-site response organizations that are generally available and that perform emergency response functions. These may include** | | X<br>The number and type of emergency services (e.g. police fire fighting services, ambulance service and medical facilities) ~~that are part of to the emergency response,~~ the emergency response training….." | | As per IAEA glossary. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| | | | | **police, firefighters and rescue brigades, ambulance services, and control teams for hazardous materials.”**<br><br>Also it is suggested to include the abovementioned definition as the footnote. | | | | |
| 87. | Iran 4-EPReSC (INRA) | Paragraph 7.66/Bullet F/Second line | “…that are part of ~~to~~ the emergency response, the emergency response…” | Editorial Comment (if not accepting the comment no.3) | | X | | Please see resolution to Iran comment 3. |
| 88. | Japan 1-EPReSC (NRA) | 7.66 (b) | (b) The size of **<u>the emergency planning zones</u>**.<br><br>Delete footnote 6. | It does not necessarily need to be limited to the urgent protective action planning zone. Reactors with power levels greater than 100 MW(th) are classified as Category I, based on GS-G-2.1. | X | | | |
| 89. | Germany 64-NUSSC | 7.82<br>Line 6 | … The operating organization should use safety assessments to inform the design of …. | Safety assessments in plural is more suitable here | X | | | |
| 90. | Pakistan 7-NSGC (PAEC) | 7.85-7.89 | Addition of new Paras may be considered regarding obsolescence of equipment and component especially I&C instrumentation in RR under ageing management or other section. | Obsolescence of equipment / component especially in I&C instrumentation is important concern in nuclear industry. So a para may be added to sensitize the designers, vendors and operating organizations to address this issue for long term safe operation of RR. | | | X | Already covered in SSG-10. |

| | | | | | | | | |
|---|---|---|---|---|---|---|---|---|
| 91. | Pakistan 8-NSGC (PAEC) | 7.85-7.89 | Addition of new Paras may be considered under ageing management for long term operation or extended operation beyond designed life. | Long term operation or extend operation beyond designed life has not been discussed in the SSR3 and DS511 | | | X | Long term operation and beyond design life is not covered as it is covered through ageing management and periodic safety reviews. |

**Section 8**

No comment

**Section 9**

No comment

**References**

No comment