**Canada's Comments on**
**DS476 Safety Requirements: Safety of Research Reactors**

| COMMENTS BY REVIEWER<br>Reviewer: M. de Vos; Majid Fassi Fehri and…<br>Country/Organization: Canadian Nuclear Safety Commission/Industry Date: Oct 9, 2015 | | | | RESOLUTION | | | |
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| Comment No. | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| 1 | 2.9 | The safety philosophy that is followed to fulfil the objectives according to the principles stated in Ref. [1] relies on the defence in depth concept and on the adoption of measures for the management and verification of safety over the entire lifetime of the research reactor facility. The safety philosophy ~~shall~~ is expected to address the means with which the organization supports individuals and groups to perform their tasks safely taking the interactions between man, technology and organizational aspects into account. | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | X The safety philosophy ~~shall~~ addresses the means… | | The text has been revised to address the comments here and below, where possible. However, the safety standards specialists indicate that there is no rule that "Shall" statements are not allowed before the first numbered requirement. For example, see paras 2.1 to 2.7 of GSR Part 3, which contain several shall statements before Req. 1 appears in para 2.7. Also see paras 2.8 and 2.10 of SSR-2/1, which contain shall statements before Req. 1. Text such as "is expected to" or "needs to" or |

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| | | | | | | | similar would weaken the paragraph and is therefore discouraged. |
| 2 | 2.12 | Application of the concept of defence in depth in the design of the research reactor provides a series of five levels of defence (based on inherent features, equipment and procedures) that are aimed at preventing accidents, and ensuring adequate protection of people and the environment against harmful effects of radiation and mitigation of the consequences in the event that prevention of accidents fails. The independent effectiveness of the different levels of defence is a necessary element of defence in depth (see para. 3.31 of Ref. [1]). However, the concept of defence ~~in depth shall~~needs to be applied with account taken of the graded approach. | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | | X | See response to comment 1 above. The concept is applied using the graded approach.. |
| 3. | 2.17 | The factors to be considered in deciding whether the application of certain requirements established here may be graded ~~shall~~ include: | Section 2 exists before Requirement 1 and therefore "Shalls" should <u>not</u> exist in any Section 2 text. | X | | | See response to comment 1 above. |
| 4 | 3.1 | GSR Part 1 requires the ~~The~~ government ~~shall~~ to ensure than an adequate legal infrastructure for a research reactor facility is available. This ~~shall~~ provides for the | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | X<br>GSR Part 1 requires the ~~The~~ government ~~shall~~ to ensure than an | | See response to comment 1 above. |

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| | | regulation of nuclear activities and for the clear assignment of responsibilities for safety in all stages in the lifetime of the facility. According to the principles quoted below the government is responsible for the adoption of legislation that assigns the prime responsibility for safety to the operating organization and establishes a regulatory body. The regulatory body is responsible for the establishment of regulations that results in a system of authorization8 for the regulatory control of nuclear activities and for the enforcement of the regulations. These principles are established in Section 3 (Principles 1, 2) of Ref. [1]. | | | adequate legal infrastructure for a research reactor facility is available. | | |
| 5 | 3.2 last sentence | The application of a graded approach that is commensurate with the potential hazards of the facility is essential ~~and shall be~~ is used in the determination and implementation of adequate safety requirements (see paras 2.15–2.17). | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | | X | See response to comment 1 above. Deleting the shall here dilutes the requirement. |
| 6 | 3.3 | GSR Part 1 requires the ~~The~~ State ~~shall~~ to establish and maintain an effectively independent regulatory body for the regulatory control of facilities and activities (Requirement 3 of Ref. [3]). To be effective, the regulatory body ~~shall~~ need to be provided with the statutory legal authority and | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | X<br>GSR Part 1 requires the ~~The~~ State ~~shall~~ to establish and maintain an effectively independent regulatory body | | See response to comment 1 above. |

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| | | resources necessary to ensure that it can fulfil its responsibilities and fulfil its functions. This includes the authority to review and assess safety related information submitted by the operating organization during the authorization process and to apply the relevant regulations (e.g. by issuing, amending or revoking authorizations or their conditions), including carrying out… | | | for the regulatory control of facilities and activities (Requirement 3 of Ref. [3]). | | |
| 7 | 3.4 | The authorization process is ongoing, starting at the site evaluation stage and continuing up to and including the release from regulatory control. The authorization process may vary among States but the major stages of the authorization process for nuclear research reactors ~~shall~~ include the: | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | | X | See response to comment 1 above. |
| 8 | 3.5 | In some cases, several stages may be authorized by a single licence, but conditions are attached to it to control the subsequent stages. Despite these differences between national practices, a detailed demonstration of safety in the form of safety analysis report which includes an adequate safety analysis ~~shall~~ is expected to be submitted by the operating organization to the regulatory body for review and assessment as part of the authorization process. | Section 2 exists before Requirement 1 and therefore "Shalls" should not exist in any Section 2 text. | | | X | See response to comment 1 above. This is an important requirement and replacing shall with "is expected" dilutes this requirement. |
| 9 | 4.17 | Those processes shall allow | "Design requirements" is | X | | | |

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| | second sentence | the operating organization to ensure that the fabrication and construction of items important to safety are performed in accordance with both ~~the~~ design requirements ~~intents~~ and ~~the~~ regulatory requirements. | more appropriate than "design intent". "Design intent" is actually formally articulated in an operating organization's design requirements (i.e. technical specifications) | | | | |
| 10 | 5 | The main safety objective in evaluating the site for a research reactor is the protection of the public and the environment against the radiological consequences of normal and accidental releases of radioactive material. NS-R-3 Site Evaluation for Nuclear Installations, Ref. [5] shall apply for evaluating the site for a research reactor. ~~(for additional requirements see Ref. [5]). Information shall be collected in sufficient detail to support the safety analysis to demonstrate that the research reactor facility can be safely operated at the proposed site. For low power research reactors, critical and subcritical assemblies the amount of detail to be provided can be substantially reduced below that required for a medium or high power research reactor (see also paras 1.6–1.9). The results of the site evaluation shall be documented and presented in sufficient detail to permit an independent assessment by the regulatory body.~~ | NS-R-3 are not "supplementary requirements": they are THE requirements to be applied to site evaluation for all nuclear facilities. NS-R-3 (and its future successor SSR-1) allow for the application of the graded approach commensurate with the characteristics of the facility.<br><br>Most section 5 requirements in this document are not necessary as they duplicate what is already captured in NS-R-3. | | X<br>The main safety objective in evaluating the site for a research reactor is the protection of the public and the protection of the environment against the radiological consequences of normal and accidental releases of radioactive material (see NS-R-3 [5]). | | It is agreed that NS-R-3 is the requirement document to apply to site evaluation. Text has been revised ("for additional requirements" deleted) to clarify NS-R-4 is not supplementary. However, it is necessary to include the remaining text as it provides important information for MS with a research reactor but without an NPP and for MS embarking on a RR project.<br>See below regarding the comment on duplication. |

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| 11 | 5.2 | Delete entire clause | Addressed in NS-R-3 already | | | X | The issue was previously discussed and the resolution was agreed to during NUSSC/ RASSC/ WASSC committee meetings. It was decided not to establish new requirements but to point to the applicable requirements in NS-R-3. This was done here in Section 5. |
| 12 | 5.4 | Delete entire clause | Addressed in NS-R-3 already | | | X | See above response to comment 11. |
| 13 | 5.5 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to comment 11. |
| 14 | 5.6 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to comment 11. |
| 15 | 5.7 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to comment 11. |
| 16 | 5.8 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to comment 11. |
| 17 | 5.9 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to comment 11. |
| 18 | 5.10 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to |

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| | | | | | | | comment 11. |
| 19 | 5.11 | Delete entire clause | Addressed in NS-R-3 already | | | X | "See above response to comment 11. |
| 20 | 6.184 | Item (b)<br><br>The entire development process, including control, testing and commissioning of design changes, shall be consistent with an instrumentation and control lifecycle approach and shall be systematically documented and shall be reviewable | Use of the lifecycle approach to computer based equipment design is now the norm across many I&C industries and forms a disciplined approach to equipment expected to become outdated quickly. | X | | | |
| 21 | 6.200 | Entire paragraph is not clearly written and needs to be revisited.<br><br>Critical and subcritical assemblies are unlikely to include spent fuel or significantly irradiated fuel and therefore the requirements related to handling and storage of spent fuel or significantly irradiated fuel may not apply. The other requirements mentioned in paras 6.195–6.198 apply. | What exactly does the text in red mean considering both critical and subcritical assemblies require something to maintain the fission reaction. | | | X | Special provisions are typically required for highly irradiated spent fuel from research reactors because of the high radiations fields. This paragraph explains that storage requirements for spent fuel may not apply to critical assemblies and subcritical assemblies. |
| 22 | 7.25 | The purpose of this requirement is not clear. Suggest deleting entire clause.<br><br>The operating organization | Why would this be a requirement? It is up to the operating organization to ensure it has suitable capabilities available to conduct its | | | X | This is in the context of Req. 69 for Operating Personnel and clarifies that the OO should arrange for |

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| | | shall arrange for the provision of assistance by contractor personnel as required. | activities in accordance with the licence. | | | | contractor personnel if technical support is needed and not available with current staff, as necessary to perform the operating functions. This is coherent with SSR2/2, 3.2 (d) |
| 23 | Appendix I I.1 | The following are examples of selected postulated initiating events for research reactors. Specific designs may have addition PIEs based on specific design characteristics: | When considering non-water cooled designs, the list provided may not be fully applicable and some PIEs may be missing. The first paragraph needs to reflect this. | X | | | Non-water-cooled reactors are out of scope. |
| 24 | Appendix I list item (7) | Under external events, add:<br><br>Electromagnetic Interference (i.e. solar events) | This is considered in NS-R-3 and is particularly important for research reactors in close confines to other high energy facilities. | X | | | |
| 25 | 1.3, Footnote 2 | A research reactor is a nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes, including experimental facilities associated with the reactor and storage, handling and treatment facilities for radioactive materials in the same building that are directly | Facilities that store, handle and treat radioactive materials from the research reactor but are outside its building are already covered by documents such as NS-R-5 and SSR-5. These facilities should not be covered by a | | | X | In this context, Site is the area under the control of the reactor management. See 5.3. It includes facilities directly related to the safe operation of the research reactor and |

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| | | related to safe operation of the research reactor. | research reactor document. | | | | under control of the reactor management. It is agreed that other facilities are covered by SSR-5. |
| 26 | 1.7 | The term covers the reactor core, radioactive sources used, experimental devices4, all systems needed for their operation, installations managed by the facility to maintain nuclear material (irradiated or not) and radioactive waste management and all other facilities relevant to either the reactor or its associated experimental facilities and devices located within the reactor building. | See comment 1. | | | X | Comment 1 in this Table is on Safety Philosophy so the reason is not clear.<br><br>See response above to comment 25 on site. |
| 27 | 2.2 (b) | To limit the likelihood of events that might lead to a loss of control over a nuclear chain reaction, the fuel cooling and the radioactive material containment; | In the original text, it was not clear what the difference between losing control of a nuclear reactor core and nuclear chain reaction, and a radioactive source and any other source of radiation was. | | | X | The text is coherent with SF-1 and links to the Fundamental Safety Principles. The suggested change would introduce inconsistencies. |
| 28 | 6.4 | In addition, the reactor design shall also consider the effects of the associated facilities on the reactor in all the stages of the reactor's lifetime (e.g. in terms of service conditions, electromagnetic fields and other interferences). | There is more risk coming from the impact of the associated facilities on the reactor (if it's impact) than the other way around. | | | X | The first sentence of 6.4 makes it clear that the design shall consider the effects of the facilities that may affect safety. Redundant. |
| 29 | Requireme | Subcritical and critical assemblies | This is to be consistent | X | | | |

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| | nt 47 footnote 34 | may not require cooling systems. | with footnote 31. | | | | |
| 30 | Instrumentation and Control Systems | Some systems may not be applicable for a subcritical or critical assembly. | Please add a footnote for this section to indicate that some systems might not be relevant to subcritical and critical assemblies, e.g. a reactor protection system for a subcritical assembly. | | | X | This is covered by the use of "Appropriate…" in the requirement. Elsewhere the use of a graded approach is recommended. |
| 31 | 7 footnote 38 | Operation includes all activities within the reactor building performed to achieve the purpose for which the nuclear research reactor was designed and constructed or modified. Besides operating the reactor, this includes: maintenance, testing and inspection; fuel handling and handling of radioactive material, including the production of radioisotopes; installation, testing and operation of experimental devices; the use of neutron beams; use of the research reactor systems for the purposes of research and development and education and training; and other associated activities. | The activities of radioisotope production or use of neutron beams, that do not have an impact because they are outside the reactor building, should not be captured by this documented as they already are with documents such as NS-R-5. | | | X | The use of neutron beams and other utilization activities are within the scope of this requirements document. |
| 32 | 7.5 and footnote 40 | In particular, in accordance with regulatory requirements, the reactor manager , the shift supervisors and the reactor operators might be required to hold an authorization (licence or certification) issued by | Remove footnote 40 and change the text as proposed given that subcritical or critical facilities' staff might not be required to have a | | | X | The use of "might be required" is discouraged and weakens the paragraph. The staff positions that |

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| | | the regulatory body, operating organization or other competent authority. | certified staff. However, their staff must undergo appropriate training. | | | | require a licence or certificate shall be determined in accordance with the legal framework of the State. |
| 33 | 4.3 | 4.3. The operating organization shall submit to the regulatory body in a timely manner any information related to the safety and licensing of the reactor that it has requested. | The term “any” opens the door for requests for any information requests. | | | X | It is reasonable to assume that the regulatory body will request only information within its mandate. |
| 34 | 6.24, 6.57, Requirement 22, 6.65, 6.67, 6.68, 6.121, 6.126,6.187 | | Throughout the document DECs are identified requiring the design to “consider” , “when practicable”, “survive beyond design basis conditions”<br>DECs are useful as part of an assessment process, they allow the determination of margin when design rules are relaxed e.g. use of Best Estimate vs worst case.<br>At the design stage, consideration of DECs would in reality result in Structures Systems & Components being designed to meet DEC not be assessed to them.<br><br>The impact is that a DEC | X | | | Thanks for the comment.<br><br>It is not clear what change is being proposed here. |

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| | | | will become the Design Basis if it is considered during the design stage of the reactor. | | | | |
| 35 | 4.11 (b) | | The IMS does not need to follow the IAEA standard. | | X<br>The relevant IAEA safety standards | | Revised to clarify the IAEA standards adopted by the state shall apply, i.e., relevant IAEA standards. |
| 36 | 6.53 | | The "short term "is not defined. The impact emergency planning and also the requirements for back up systems. | | | X | The text is consistent with SSR-2/1 which was approved. |
| 37 | 6.60 | | The term "prompt" needs clarification so that analysis assumptions can be agreed. The result is that automatic systems may not be identified for events which require them if prompt is taken to be a longer period. If its taken to be a shorter period than intended, safety is likely not adversely impacted but project costs and time for implementation may be. | | | X | The text is consistent with SSR-2/1. |
| 38 | 6.112 | | Is obsolescence in this context related to the aging management and not design. It is unclear how the design of a new | | | X | The text was revised to include "obsolescence" in response to Canada's comment |

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| | | | research reactor would consider obsolescence | | | | #77 during the MS review at STEP 10 |
| 39 | Requirement 38 | | Potential additional point for consideration<br>Approaches to critical following complete defuel can result in normal monitoring instruments being off scale as fields very low. | | | | Thanks for the comment it was carefully considered.<br>After a complete defuel, the approach to critical is usually arranged according to the sequence for commissioning. The requirements are covered in Section 9. |
| 40 | 3.1 | The government shall ensure that an adequate legal infrastructure for a research reactor facility is available. | Spelling<br>Replace “than” with “that” | X | | | |
| 41 | 4.17 | During the manufacturing and construction of systems, structures and components of the research reactor, including its | Spelling<br>Change to “components” | X | | | |

**TITLE: DS476 Safety of Research Reactors (Sept 2015)**

| COMMENTS BY REVIEWER<br>Country/Organization: FRANCE Date:<br>pages | | | | RESOLUTION | | | |
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| Comment No. | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| 1 | General comment: | The main aspects of research reactors are eclipsed or not enough underlined or highlighted (besides the document is not synthetic, not easy readable):<br>• in the context of a complete revision of the NS-R-4, it would have been useful if the new document could really **highlight** the **safety specificities of research reactors (mainly requirements related to the graded approach and to experimental devices and interactions between them and the reactor)** considering of course that all general safety requirements defined by the AIEA are applicable to research reactors (NS-R63, GSR Part 3, GSR Part 4…).<br>For example, experimental devices are more clearly mentioned in "scope" and other chapter of the DPP<br>• It appears that there are too many general requirements that are non-specific to research reactors but also applicable to others nuclear facilities.<br>At least, specificities of research reactors should be **clearly identified**. | | | | | This general comment has been carefully considered. It is felt that the specific requirements for research reactors are clearly identified. |
| 2 | 2.12 | Paragraph 3.31 of the Safety Fundamentals [1] states that "the independent effectiveness of the different levels of defence is a necessary element of defence in depth" ~~(see para. 3.31 of Ref. [1])~~. | To be fully consistent with SSR-2/1 that mention a quotation of SF1. Indeed, "necessary" could be understood in a document related to principles, but this word is not anymore relevant for a requirement | X | | | |

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| 3 | 1.7 | This definition excludes nuclear reactors used for the production electricity, naval propulsion […] heating. It includes nuclear reactors used for research and demonstration purposes and connected to the electrical network. | The case of research reactors connected to the electrical network is not clear (Astrid), as well as demonstration reactor for naval propulsion (RES). | | | X | It is agreed that the scope of this document does not include reactors used for the production of electricity or for naval propulsion. Sodium cooled fast reactors (Astrid) are also out of scope. Such reactors typically have power levels in excess of several tens of megawatts. For such facilities, the requirements (and engineering standards) to be applied, the extent of their application and any additional safety measures that may need to be taken are required to be proposed by the operating organization and to be subject to approval by the regulatory body. (See 1.8) |

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| 4 | 2.5 | ~~*Nuclear or radiation*~~ *risks / accidents* or *nuclear, chemical and radiation risks/accidents* | The principle considers only "nuclear or radiation risks and accident". For some research reactors, other risks such as chemical have to be considered | | | X | It is agreed that other risks such as chemical are important. However paragraph 2.5 explains how the requirements are derived from SF-1 and therefore, for coherence, it retains the same principles which focus on nuclear or radiation risks and accidents. See para 1.10 re exclusion of conventional industrial safety and non-radiological impacts. |
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# TITLE Comments on DS476 "Safety of Research Reactors"

| COMMENTS BY REVIEWER<br>Reviewer: Japan NUSSC<br>Country/Organization: Japan/NRA | | | Page 1 of 1<br>Date: 9 Oct. 2015 | RESOLUTION | | | |
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| No. | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modif./reject. |
| 1 | Req.22 | Describe some examples of DEC in a footnote such as follows;<br>● Significant fuel degradation due to loss of coolant<br>● Significant irradiated fuel degradation due to uncovering coolant in the pool | In the discussion of DS478 "Safety of Nuclear Fuel Cycle Facilities" at the 39th NUSSC meeting, it was agreed to add some DEC examples in order to better understand DEC concept of fuel cycle facilities. Also it is recommended to describe DEC example for research reactors in DS476. | | | X | Adding such examples is discouraged in a high level requirements documents such as this safety standard; it is considered more suitable for guidance documents. It also understood that these examples will not be included in DS478. |
| 2 | 7.121 | On the basis of the results of the periodic safety review, the operating organization shall implement any necessary corrective actions and shall consider making justified modifications to enhance safety (see also para. 7.120 ~~7.119~~ on the interaction between ageing management and periodic safety reviews). | Editorial. | X | | | |

**DS476, Safety of Research Reactors (Step 11)**

| COMMENTS BY REVIEWER<br>Reviewer: Department of Instrument, Control, and Electrical System, Department of Safety Evaluation<br>Country/Organization: Republic of Korea / Korea Institute of Nuclear Safety<br>Date: October 9, 2015 | | | | RESOLUTION | | | |
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| Comment No. | Para/Line No. | Identified problem/Proposed new text | Reason/Description | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| 1 | Page 8<br>§2.11 | 2.11. Application of the concept of defence in depth throughout design and operation provides protection against transients, anticipated operational occurrences and accidents, including those resulting from equipment failure or inappropriate human action within the installation and events induced by external hazards. | To provide clear understanding, the "human action" should be replaced with "inappropriate human action." | X | | | |
| 2 | Page 55<br>§6.140 | 6.140. All foreseeable reactor core configurations, including the initial core through to the equilibrium core, as appropriate, shall be considered in the core design. The effect of the inserted experimental devices or irradiating materials shall also be considered. For subcritical assemblies this includes assurance that all these configurations are subcritical with justified margins. | In research reactors, the roles of experimental devices and irradiation of test materials are important. Therefore, the effect of the inserted experimental devices or irradiating materials are added. | | X<br>The effect of the inserted experimental devices or materials under irradiation shall also be considered. | | Revised for Clarity |
| 3 | Page 59<br>Req. 49 | *The paragraphs relevant to the Requirement 49 should be changed or added including means of communications (see Requirement 29 in Ref. [7] (SSR-2/1)).* | It is generally recognized that the communication system is very important for the safe operation of research reactor during accidents or emergency conditions. | | | X | It is agreed that effective means of communication is important for safety during an emergency. This item is already |

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| | | | | | | | covered in Req. 32. It is also covered in 6.91, 6.185, 6.189 and the footnote. Inclusion in Req 49 would be redundant as Req. 49 covers Provision of instrumentation and control systems for a research reactor facility |
| 4 | Page 62 Req. 52 | Requirement 52: Use of computer based equipment in systems important to safety for a research reactor<br><br>If a system important to … an integrated management system. Computer based important system and communication/ and networks systems including reactor protection system are adequately protected against cyber attacks, up to and including the design basis threat.<br><br>The SDOE(Secure Development and Operational Environment) to ensure the high functional reliability of Computer based important system is established and maintained during lifetime of digital computer system. | New texts should be added because the cyber security and SDOE are to secure high reliability. | | X (computer based system and communication and networks systems important to safety including reactor protection systems, are to be adequately protected against cyber attacks, up to and including the design basis threat) | | Text added to 6.184 (f) as revised for clarity. |
| 5 | Page 64, Req. 56 | *The paragraphs relevant to the Requirement 56 should be added* | Independent emergency power supply system should be provided for | | | X | Req 56 and Para 6.192 include |

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| | | *emergency power supply provisions (see Requirement 68 in Ref. [7](SSR-2/1)).* | the safe operation of research reactor during accident conditions. | | | | provisions for emergency power supply:<br>The design for a research reactor facility shall …consider reliable emergency electrical power supply systems.<br>In the design basis for the emergency power supply, due account shall be taken of the postulated initiating events and the associated safety functions to be performed, to determine the requirements for capability, availability, duration of the required power supply, capacity and continuity. |

**Comments on "Safety of Research Reactors"**
**(Draft Specific Safety Requirements to supersede Safety Standard Series No. NS-R-4) (DS476)**

| COMMENTS BY REVIEWER<br><br>Reviewer: U.S. Nuclear Regulatory Commission<br>Country/Organization: USA / NUSSC Date: 14 October 2015 | | | | RESOLUTION | | | |
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| Comment No. / Reviewer | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| 1 | Footnote 37 | Emergency response facilities and locations are addressed in Ref. [10]. A facility or defined area within a facility should be designated as an emergency support center from which emergency control directions will be given. The support center should be located to oversee operations effectively, but should be separated from actual activities to function efficiently. | For research reactors, there is no need for a technical support center, operational support center. There should only be a designated emergency support center. | | X<br>Emergency response facilities and locations are addressed in GSR Part 7 [6]. For research reactors, emergency response facilities (which are separate from the control room and the supplementary control room) include the emergency centre, and the technical support centre and the operational support centre, as appropriate | | It is true that many research reactors not need a technical support centre and an operational support center. However, some high power research reactors have a technical support centre and an operational support centre.<br>The Footnote text has been revised to address your concern by indicating that the emergency response facilities include the technical support and the operational support centre, as appropriate. The use of normative text "should be" for the |

| COMMENTS BY REVIEWER<br><br>Reviewer: U.S. Nuclear Regulatory Commission<br>Country/Organization: USA / NUSSC Date: 14 October 2015 | | | | RESOLUTION | | | |
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| Comment No. / Reviewer | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| | | | | | | | area and location is discouraged and therefore not included in the footnote. |
| 2 | 6.189 | Information about ~~important reactor parameters and radiological conditions at the reactor facility and the site~~ monitoring systems and laboratory facilities that are to be used to determine the need to initiate emergency measures, as well as those to be used for continuing assessment, shall be provided to the relevant emergency response facilities. These monitoring systems may consist of equipment such as radiological monitors, sampling equipment, earthquake sensors, fire and combustion product detectors, and process monitors that provide pertinent facility system or status information. Each facility shall be provided with means of communication | Expanding on the description of monitoring equipment consistent with ANSI/ANS-15.16-2015. | | X<br>Information about important reactor parameters and radiological conditions at the reactor facility and the site, and about monitoring systems and laboratory facilities that are to be used to determine the need to initiate emergency | | It is necessary to retain information about important reactor parameters and radiological conditions at the reactor facility and the site.<br><br>The list of examples of the types of monitoring equipment is more suitable for guidance. |

| COMMENTS BY REVIEWER<br><br>Reviewer: U.S. Nuclear Regulatory Commission<br>Country/Organization: USA / NUSSC Date: 14 October 2015 | | | | RESOLUTION | | | |
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| Comment No. / Reviewer | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| | | with the control room, the supplementary control room and other important locations at the facility, and with on-site and off-site emergency response organizations. | | | measures, as well as those to be used for continuing assessment,… | | |
| 3 | Requirement 81 | The operating organization for a research reactor shall prepare emergency arrangements for preparedness for, ~~and~~ response to, and recovery from, a nuclear or radiological emergency. | Including recovery adds value to this document, as IAEA documents generally do not address recovery actions. | | | X | The text is retained for consistency with Principle 9 of the Safety Fundamentals and Requirement 18 of the SSR/2-2. |

| COMMENTS BY REVIEWER<br>Reviewer: U.S. NRC<br>Country/Organization: USA<br>Date: April 27, 2015 | | | | RESOLUTION | | | |
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| 1 | multiple | Eliminate (delete) all reference to subcritical assemblies from this document and limit the scope of the body of the document to research reactors exclusively.<br><br>Proposed approach: Develop a focused appendix to gather the pertinent requirements for subcritical assemblies and have the appendix reviewed by specialists who are experts in the design and regulation of these facilities. | Generally, there are 2 categories of subcritical assemblies - those that utilize natural uranium fuel with a light water moderator and those that use enriched uranium that may or may not use more exotic moderating materials such as heavy water. The hazards associated with the natural uranium fuel and light water moderated subcritical assemblies are so minimal that their inclusion in this document is not warranted from a safety perspective.<br>In the case of subcritical assemblies using enriched uranium, criticality control is a very important safety consideration. As such, there are concerns with the level of relevant operational experience with these higher performance subcritical assemblies possessed by technical contributors to this document. Many years of research reactor operating experience supports the research reactor guidance contained in the document; however, due to the relative | | | ✓ | The scope of the document includes subcritical assemblies, therefore this cannot be deleted.<br>IEXs with experience in subcritical assemblies participated and contributed to the development of this doc. |

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| | | | rarity of high performance subcritical assemblies, an equivalent level of expertise does not support the guidance related to the subcritical assembly. The concern is that critical design and operating guidance specific to the high performance subcritical assemblies may have been overlooked and not included in this document. For this reason, it is important to develop a separate document related specifically to the safety of high performance subcritical assemblies drafted by technical contributors with significant design and operational experience with these facilities. | | | | |
| 2 | multiple | With respect to the multiple references to the design extension conditions, the document needs to recognize, through the addition of appropriate language related to the application of a graded approach, that smaller research reactors (typically less than 2 $MW_t$) may not have the same vulnerability to design extension conditions and do not require additional assessment | The need for significant assessment of design extension conditions and additional mitigating actions may not be necessary for all research reactors. The reader needs to be aware that under some conditions additional assessment and actions may not be needed. The need for action must be determined by the assessment of the hazards presented by a research reactor for each applicable design | | ✓ | ✓ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 2) to address the comments on design extension conditions. We agree that the need for action must be determined by the assessment of the |

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| | | and mitigating actions.<br>Proposed resolution:<br>Delete the concept of design extension conditions from the main body of the requirements document (Note: This is consistent with the IAEA Task Force review of the requirements documents, i.e., no changes were necessary for NS-R-4.) Write an annex to the document that highlights:<br>1) Design Extension Conditions apply to higher power research reactors, per paragraph 1.8.<br>2) The design of research reactors should consider operational history, per revised para. 6.37.<br>3) Important Fukushima considerations for research reactors | extension event and then compared to the State's criteria for unacceptable radiological consequences. | | | | hazards presented by a research reactor; this is covered in the text. |
| 3 | 1.3 | | Section 1.3 includes both critical assemblies and subcritical assemblies within the definition of a research reactor, but Footnote 2 only mentions critical assemblies. This seems | ✓ Added both to footnote 2 | | | |

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| | | | inconsistent. | | | | |
| 4 | 1.9 | | Section 1.9 contains the term "research reactors or subcritical assembly," whereas above a research reactor was defined as including subcritical assemblies. A general comment (consistent with the preceding comments) is therefore that the scope and clarity of such terms throughout the document appears to vary, and care should be taken that the terms are used consistently. | ✓ Document checked and revised for consistency. | | | |
| 5 | 1.3 | In footnote #2, consider defining critical and subcritical assemblies similar to how the term research reactor was defined.<br>Recommend separating subcritical assemblies from DG476 because reference to these facilities is disjointed and the document is not fully applicable to these facilities | | | ✓ Footnote 2 revised to include critical and subcritical assemblies | | The document includes subcritical assemblies in the scope therefore it is not appropriate to separate. |
| 6 | 1.8 | Consider using a separate document to address homogenous reactor or accelerator driven systems. | | ✓ | | | Considered. This may be addressed elsewhere. |
| 7 | 2.8 | Remove text "safety features for design extension conditions."<br>See proposed resolution of comment | 1. Concept generally does not apply to the range of research reactors because of their small risk. | | ✓ | ✓ | See the resolution to comment #28: The text has been revised |

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| | | #2 on DECs. | 2. The concept is too vague to be layered upon the concept of "graded approach."<br>3. See proposed change to para. 6.37.<br>4. The IAEA review of requirements documents for Fukushima did not identify this as a concern.<br>5. Design extension conditions apply to a small segment of high power research reactors, per paragraph 1.8. | | | | in Para 6.64 (Req 22) to address the comments on design extension conditions.. |
| 8 | 2.12 | | The application of the five-layer defense-in-depth structure to critical and subcritical assemblies is not readily apparent, and may be different than for other research reactors. This has been a topic of much discussion with regard to criticality safety at fuel facilities, where many of these concepts don't seem to fit. As an example, item (3) states that engineered controls shall be capable of transferring the research reactor first to a controlled and then to a safe state, but for a subcritical assembly in particular, there's no such thing as a | | | ✓ | It is recognized that the application of DiD may be different for subcritical assemblies than for other research reactors. However, this is addressed in para. 2.12 where it states that the concept shall be applied with account taken of the graded approach. |

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| | | | controlled state. Safety consists in keeping it subcritical. | | | | |
| 9 | 2.12 | | Clarify the requirement for containment vs. confinement functions | | ✓ | | This is clarified in footnote 26 and in the Glossary |
| 10 | 2.14 | Delete item (4), referring to design extension conditions. | 1. Concept generally does not apply to the range of research reactors because of their small risk.<br>2. The concept is too vague to be layered upon the concept of "graded approach."<br>3. See proposed change to para. 6.37.<br>4. The IAEA review of requirements documents for Fukushima did not identify this as a concern.<br>5. Design extension conditions apply to a small segment of high power research reactors, per paragraph 1.8. | | ✓ | ~~✓~~ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the comments on design extension conditions. |
| 11 | 3.2 | | Clarify the term, "global safety regime." | | ✓ | | The term is clarified in GSR Part 1, Ref. [3]. |
| 12 | 4.6/1 | The safety policy of the operating organization shall include a commitment to achiev~~eing enhancements~~ and maintain~~in~~ | The regulatory body shall define the minimum level of safety through their regulatory framework, and an expectation of positive safety culture. | | | ✓ | The text is coherent with SSR-2/2, 4.5. Continuous enhancement is |

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| | | compliance with all regulatory requirements established by the regulatory body to ensure operational safety. ~~The strategy of the operating organization for enhancing safety and for finding more effective ways of applying and, where feasible, improving existing standards shall be continuously monitored, periodically revised and supported by means of a clearly specified programme with clear objectives and targets.~~ | The operator must achieve and maintain compliance with the regulatory body's requirements. To impose an expectation on the operator to continuously enhance the safety at their facility is an unreasonable expectation and likely very difficult for a small facility, with a small operating staff, and limited funding. | | | | consistent with a safety culture that discourages complacency and encourages a questioning and learning attitude. |
| 13 | Requirement 4 | Requirement 4: Integrated management system for a research reactor facility<br>The operating organization for a research reactor facility shall establish, implement, assess and ~~continuously~~ improve ~~an~~ the facility's integrated management system as necessary to ensure facility safety. | The regulatory body shall define the minimum level of safety through their regulatory framework. The operator must achieve and maintain compliance with the regulatory body's requirements through an integrated management system. To impose an expectation on the operator to continuously improve the integrated management system absent a potential adverse impact on facility safety the safety at their facility is an unreasonable expectation and likely | | | ✓ | See above resolution to comment #12. |

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| | | | very difficult for a small facility, with a small operating staff, and limited funding. | | | | |
| 14 | 4.7 | | Clarify whether the term 'governed by the potential hazard of the reactor' is the same as 'graded approach.' | | | ✓ | This is applied using the graded approach. |
| 15 | 4.20, footnote 15, pg. 20 | "…or by independent external ~~independent~~ organizations." | Editorial | ✓ | | | Covered in footnote 12 |
| 16 | 4.23 | "Deterministic safety analysis shall be the primary tool for safety assessment of research reactors. Probabilistic safety analysis may be used as a complementary tool for improving the safety assessment." [no change] | We strongly agree with this statement. | ✓ | | | |
| 17 | 5.5/1 | 5.5. If the evaluation of the site and the operations area for these *four factors*, including their foreseeable evolution,…….. | It is not clear as to what the four factors are referred to in section 5.5. In the Step 7 version of the document, Section 5.4 listed 4 aspects to be considered. In the Step 8 version of the document there are now 6 aspects to be considered. Section 5.5 appears to refer to the aspects for consideration provided in Section 5.4. If so, then the number needs to be changed from 4 to 6 and the reference to "aspects" in | ✓ | | | 5.5 changed to 6 aspects for consistency. |

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| | | | Section 5.4 and "factors" as used in 5.5 needs to be consistent between the two Sections. | | | | |
| 18 | 5.7 | Delete "and postulated worst combination of low probability but high consequence events that may exceed those conditions assumed in the design basis accident resulting in design extension conditions." | 1. This is a new requirement beyond Ref. 5.<br>2. Research reactors did not require any modifications based on IAEA Fukushima Task Force review. | | ✓ … and postulated worst combinations of low probability but high consequence events that may exceed those conditions assumed for design basis accidents shall be collected for the region in which the potential facility site is located … | ~~✓~~ | |
| 19 | 5.8 | "…it shall be confirmed that ~~there will be no insurmountable~~ | Clarify the term "insurmountable difficulties." | ✓ | | | |

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| | | ~~difficulties in the development of~~ off-site emergency arrangements, where appropriate, will be available prior to the start of reactor operation…" | | | | | |
| 20 | 5.12 | "…the suitability of the site to accommodate a nuclear installation shall be carefully analysed to ensure agreement with regulations related to ~~avoid unacceptable~~ radiological risk to site personnel and public." | Clarify the phrase, "unacceptable radiological risk to site personnel and public." | | ✓ ensure agreement with regulations related to… | | Text also revised to address comments from USA, GER and IRA |
| 21 | 6.8 | Delete "or large radioactive releases are practically eliminated."<br>Delete footnote 22. | This is a power reactor concept that does not apply to research reactors except per paragraph 1.8. It only adds complexity, without substantial benefit. | | | ✓ | This applies to medium and high power research reactors. |
| 22 | 6.17 | Delete "In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall be as far as practicable independent of safety systems." | This is a power reactor concept that does not apply to research reactors except per paragraph 1.8. It only adds complexity, not clarity. | | ✓ | ~~✓~~ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the comments on design extension conditions |
| 23 | 6.24 / 3 | "…reactor ~~and selected design extension conditions~~ shall be identified…" | This is a power reactor concept that does not apply to research reactors except per paragraph 1.8. It only adds complexity, not clarity. | | ✓ | ~~✓~~ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) |

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| | | | | | | | to address the comments on design extension conditions |
| 24 | 6.37 | Add the following sentence to the end of the requirement:<br>"This includes consideration of events or conditions that are beyond those design practices of current research reactors, similar to the concept of design extension conditions for nuclear power plants." | This is a more practical and implementable design consideration for research reactors as compared to the wholesale imposition of the power reactor concept of design extension conditions. | | ✓ | ✓ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the comments on design extension conditions |
| 25 | 6.49 | | It is difficult to conceive of how the addition of moderator from firefighting systems would not "increase the criticality risk" for a subcritical assembly. A more reasonable criterion would be to design subcritical assemblies to be safely subcritical when optimally or fully flooded. | | ✓ | | Text added (footnote 25) to indicate that subcritical assemblies shall be designed to be safely subcritical when fully flooded. |
| 26 | 6.62/2 | 6.62. The design of subcritical assemblies shall include technical provisions to prevent ~~inadvertent~~ criticality. ~~conditions~~ (see para. 6.66). | In a subcritical assembly, criticality is to be prevented under all circumstances. Section 6.66 confirms that intent | ✓ | | | |

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| 27 | Requirement 22 | Delete Requirement and associated sub-paragraph.<br>Write an annex to the document that highlights:<br>1. Design Extension Conditions apply to higher power research reactors, per paragraph 1.8.<br>2. The design of research reactors should consider operational history, per revised para. 6.37.<br>3. Important Fukushima considerations for research reactors | Deleting the concept of design extension conditions from the main body of the requirements document is consistent with the IAEA Task Force review of the requirements documents, i.e., no changes were necessary for NS-R-4. | | ✓ | ✓ | See revision of text to address comm 28. |
| 28 | 6.64/1 | 6.64. An analysis of design extension conditions shall be performed to determine if the potential radiological consequences exceed those deemed unacceptable by the State. The main technical……. | If the postulated radiological consequences exceed those the State has determined to be unacceptable, then some type of action is necessary. The tone of this paragraph (6.64) leads the reader to believe the only acceptable options are a revised design | | ✓ An analysis of design extension conditions shall be performed[1] to determine | ✓ | The text has been revised to further clarify that the analysis of design extension conditions is to determine whether the potential radiological |

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[1] The analysis of design extension conditions could be performed by means of a best estimate approach (more stringent approaches may be used according to States' requirements).

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| | | | or extended capability of the safety system. It ignores the option for the inclusion of additional mitigative strategies using portable equipment and/or operator actions. The ignored option is likely to be the most reasonable option for existing research reactors. | | whether the potential radiological consequences would exceed those deemed unacceptable by the relevant authority. | | consequences would exceed those deemed unacceptable by the relevant authority (State). |
| 29 | 6.66 | | The term "inherent safety provisions" is vague and not commonly used in the industry. Would this include fixed assembly geometry, spacing, and the use of fixed neutron absorbers? Limiting the assembly to natural uranium or a limited quantity of fissile material may not be feasible, and reliance on these other parameters may be necessary.<br>The meaning of "mitigatory measures" is also unclear. Is this mitigation in the sense of limiting the resulting dose if criticality cannot be sufficiently precluded (e.g., by shielding and/or | | ✓<br>"inherent" deleted. | | Additional text added for clarification. Measures for mitigating the consequences shall be determined and implemented on the basis of safety analysis. |

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| | | | remote assembly)? The meaning should be clarified. | | | | |
| 30 | 6.91/2 | The availability of reliable and diverse means of communication necessary for safety and emergency response within the reactor facility, including the supplementary control room (if there is one), and with the emergency centre, shall be ensured at all times... | Footnote 28 should be included in the text to emphasize its importance as a design criterion, and not relegated to a footnote.<br><br>Communication with the emergency centre should be added, consistent with paragraph 6.185.<br>The necessity of the communications should include the purpose of "emergency response" consistent with the purpose statement of Requirement 32. | | | ✓ | The text is coherent with SSR-2/1, 5.66 and 5.67. Additional requirements here would introduce redundancy with 6.185 and FN 28. Communication systems for emergency response are also covered in 7.93. |
| 31 | 6.121 | Delete (e) and "and the safety features for DEC" in part (g). | 1. The concept generally does not apply well to research reactors.<br>2. See comment on para. 6.37.<br>3. Not identified in IAEA Fukushima review. | | ✓ | ✓ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the comments on design extension conditions |
| 32 | 6.128 / 7 | Delete, "…and, to the extent practicable, in design extension conditions" | The graded approach and "to the extent practical" are overlapping and confusing concepts. | | ✓ | ✓ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the comments on design |

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| | | | | | | | extension conditions. |
| 33 | 6.131 | Delete 2nd sentence:<br>The resistance of barriers in design extension conditions shall be analysed for determination of adequacy considering planned mitigation measures. | This is a new and complex requirement that goes beyond even the power reactor requirements. | ✓ | | | |
| 34 | Requirement 44 | Delete "and where appropriate design extension conditions." | This is a new, unwarranted requirement that goes beyond even the power reactor requirements. | ✓ | | | Coherent with SSR-2/1. |
| 35 | 6.140 | "All foreseeable intentional or abnormal reactor core configurations…" | Does the phrase "all foreseeable reactor core configurations" include those arrived at through abnormal conditions, such as by a fuel misload condition (as occurred in the 1983 accident)? This requirement should be applied in accordance with the double contingency principle, such that no single fuel misload can lead to criticality. It is interesting that the entire document does not mention such a fundamental principle of criticality safety (i.e., double contingency). | | . | ✓ | "All foreseeable" covers all operating states and design basis accidents conditions. This includes intentional or abnormal conditions. Double contingency is commonly applied to fuel cycle facilities. |
| 36 | 6.143, footnote 30 | | Consider defining "critical facility." | | ✓ critical assembly | | Changed to critical assembly. |

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| 37 | 6.145 | "…criticality cannot be reached ~~in any~~ by any single change in the core configuration…" | The requirement stated here, that criticality in a subcritical assembly cannot be reached in any core configuration, temperature, moderation, or reflection conditions, seems to go beyond what is normally required for fuel applications, namely the double contingency principle. It is certainly possible to achieve criticality as long as greater than a critical mass is present. What is normally required in fuel applications is that the assembly must be shown to be subcritical following any single change in process conditions. Clarity would be achieved by putting this section in the context of meeting double contingency. | | | ✓ | All planned core configurations are to be analyzed. The requirement for a subcritical assembly is that critically cannot be reached in any core configuration. A single change is a subset which is also covered here. See also NS-G-4.3 |
| 38 | 6.147 | | Same comment as for 6.145 and 6.140. "The subcritical condition shall be justified for any configurations" is too broad and not consistent with the historical approach to criticality safety (double contingency). | | | ✓ | This is reasonable given the context of any core configuration for subcritical assemblies. |
| 39 | 6.149 | | Same comment as for 6.147, 6.145, and 6.140. In addition, "may not be required" should be changed to "are | | | ✓ | "may" is suitable as the designer or operator may choose |

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| | | | not required," because it is never necessary to control reactivity if criticality is not possible. | | | | to include this provision. |
| 40 | 6.150 | | Footnote 31 to 6.150 does not make any sense. A subcritical assembly does not need to be "shut down," as it is already subcritical. The term "shut down" is normally taken to mean that it is rendered subcritical, but in this case, removing the neutron source will only reduce the fission rate which, for a critical assembly, is generally referred to as the power level. Either this section should refer to critical, and not subcritical, assemblies, or else the meaning of the term "shut down" should be explained. | | ✓<br>Add quote "shutdown" to indicate no longer in operation mode | | The term "shutdown" is used here to indicate that the subcritical assembly is no longer in operation mode. |
| 41 | 6.155 | Delete "and in design extension conditions without core melt" | This is a new requirement that is overly complex and prescriptive. | | ✓ | | Text revised to accident conditions for coherence with SSR-2/1 and Req. 46 |
| 42 | 6.162 | | Section 6.162 states that subcritical assemblies do not require cooling systems, but this disagrees with Footnote 30, which states that some | | ✓<br>For heat removal | | Clarification add to show cooling not required for heat removal, but provisions shall be |

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| | | | critical and subcritical assemblies may not need cooling. Footnote 33 also says they do not need cooling systems, but Footnote 35 says they may not need emergency cooling systems. It is correct to say that subcritical assemblies do not require cooling for heat removal to prevent damaging the fuel (although conceivably cooling could be provided for other reasons). However, it is confusing to then say that, although they do not require cooling systems, they still need to monitor and control the coolant, etc. If there is no cooling system, there is no coolant. | | | | provided in fluid systems to preserve the fuel elements and SSC. Footnote also clarified. |
| 43 | 6.164 | Delete 2nd sentence:<br>"Special procedures for cooling the core shall be considered in the case of selected design extension conditions." | This is redundant to the graded approach. | | ✓ | ✓ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the comments on design extension conditions t is appropriate to consider special procedures. Important lesson |

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| | | | | | | | learned from FD. |
| 44 | 6.202/3 | Where liquid ~~(and gaseous)~~ radioactive waste is to be handled, provision shall be made for the detection of leakage and the recovery of waste, if appropriate. Where gaseous radioactive waste is to be handled, provisions shall be made for the detection of leakage and to prevent or control its release to below State established environmental release limits. | It is not clear how, from a practical perspective, one would recover gaseous waste. Once leaked, one may be able prevent release via facility isolation, or one can delay its release allowing for radioactive decay, or one can release it, if within regulatory release limits. | | ✓ …below release limits. | | Agreed. Further simplification …below release limits.. |
| 45 | 6.210/6 | (c) The facility layout permits safe movement of the lifting equipment and of items being transported~~;~~ in accordance with analyzed safe load pathways; | Pre-established safe load pathways can significantly reduce the risk of damage to SSC important to safety from lifting equipment failure or mishandling. | ✓ | | | The text is consistent with SSR-2/1. |
| 46 | 6.210/10 (new) | (e) lifting equipment can be inspected on a periodic basis | It is essential, as a matter of industrial and nuclear safety, that lifting equipment is inspected routinely. The design of this equipment should facilitate such inspection. | ✓ | | | |
| 47 | Requirement 66/7 | ……will not compromise confinement or will not lead to ~~an~~ unacceptable ~~radiation exposure~~ | There are other consequences other than radiation exposure (e.g; contamination of the environment) | ✓ | | | |

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| | | radiological consequences. | | | | | |
| 48 | Requirement 72 | Requirement 72: Performance of ~~safety related~~ activities ~~for~~ at a research reactor facility<br>The operating organization for a research reactor facility shall ensure that ~~safety related~~ activities are adequately analyzed and controlled to ensure that the risks associated with harmful effects of ionizing radiation are kept as low as reasonably achievable. | Not all activities performed in research reactor radiations areas are safety related; therefore, all activities performed in radiation areas at research reactors should be considered. The terminology used should not make unnecessary reference to safety classification schemes. | | | ✓ | Yes the Operating Organization should analyze all activities but the scope of Requirement 72 is appropriately focused on safety related activities. Others general activities are covered, for example, in Sec. **4**. |
| 49 | 7.52 | | Section 7.52, together with Footnote 41, make it clear that some commissioning testing may not be needed for subcritical assemblies. This is correct, but the equivalent of "initial criticality tests," namely verifying adequate subcriticality (e.g., through 1/M calculations) is appropriate and advisable. | ✓ | | | Suggested text added to footnote for clarity. |
| 50 | 7.58 | In item (g), delete, "and, to the extent feasible, to design extension conditions" | This is not a practical concept for research reactors. | | ✓ | ✓ | See the resolution to comment #28: The text has been revised in Para 6.64 (Req 22) to address the |

| COMMENTS BY REVIEWER<br>Reviewer: U.S. NRC<br>Country/Organization: USA<br>Date: April 27, 2015 | | | | RESOLUTION | | | |
|---|---|---|---|---|---|---|---|
| Comment No. / Reviewer | Para/Line No. | Proposed new text | Reason | Accepted | Accepted, but modified as follows | Rejected | Reason for modification/rejection |
| | | | | | | | comments on design extension conditions |
| 51 | New requirement | | This document should also include a requirement for the financial qualification of the operator that demonstrated sufficient financial assurance that the research reactor can be safely designed, constructed, operated and decommissioned. | | | ✓ | This is covered in the Code of Conduct on the Safety of Research Reactors. |
| | | | | | | | |