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| **From:** | Holbrook, Mark R |
| **To:** | AdvancedRxDCComments Resource |
| **Cc:** | Jim C Kinsey; George Flanagan; Belles, Randy; Poore III, Willis P.; Tanju Sofu; Wayne Moe; David Alberstein; Tom King; Fred Silady |
| **Subject:** | [External_Sender] DOE Laboratory Team Comments on the NRC"s Draft Advanced Reactor Design Criteria |
| **Date:** | Friday, June 03, 2016 5:29:21 PM |
| **Attachments:** | DOE Lab Team - NRC ARDC Comments.pdf<br>DOE Lab Team - NRC SFR-DC Comments.pdf<br>DOE Lab Team - NRC HTGR-DC Comments.pdf |

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The DOE national laboratory team appreciates the NRC staff's consideration of the inputs previously provided during Phase 1 of this joint DOE-NRC initiative when developing this initial draft set of non-LWR design criteria for public comment. Our comments and related feedback are provided in three attachments to this email as follows:

- Comments on the draft Advanced Reactor Design Criteria, responses to the series of NRC additional questions, and questions/comments on the public review document's introductory text are provided in a file named "DOE Lab Team - NRC ARDC Comments.pdf"
- Comments on the draft Sodium Fast Reactor Design Criteria are provided in a file named "DOE Lab Team - NRC SFR-DC Comments.pdf"
- Comments on the draft modular High Temperature Gas Cooled Reactor Design Criteria are provided in a file named "DOE Lab Team - NRC HTGR-DC Comments.pdf"

We look forward to the continued engagement of the NRC staff, and to future industry stakeholder interactions through the Regulatory Guide public meeting and comment processes, as these draft design criteria are further updated and refined.

Please contact us if you have questions or require clariifyng information regarding any of our comments and observations.

Regards,
Jim Kinsey, Regulatory Affairs Director
Idaho National Laboratory
Jim.Kinsey@inl.gov
208-569-6751 (cell)

**Mark Holbrook - Distinguished R&D Engineer**
**Advanced Reactor Technologies (ART) Regulatory Affairs**
**Idaho National Laboratory**
**(208) 526-4362 (Office)**
**(208) 351-9858 (Cell)**
**mark.holbrook@inl.gov**

| DOE Laboratory Team Comments on NRC Invitation and Instructions | | |
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| **Section** | **NRC Language** | **Team Comments** |
| **Process** | While developing the final RG, the NRC intends to consider the extent to which risk-informing the ARDC, SFR-DC, and mHTGR-DC is possible given the level of design information and data available. | The scope and intent of this statement within the context of this joint DOE-NRC initiative is not clear, and raises a series of questions, including:<br>• Does "consider" mean that risk-informing the current draft set of criteria will be factored into the final RG?<br>• How may NRC's considerations in this area affect the content of the current draft guidance that has been provided for comment?<br>• If the final RG will be affected, will stakeholders be provided with an opportunity to participate?<br>• How will NRC's considerations in this area affect the timeline for issuance of the intended Regulatory Guide?<br>• What are the implications of such an effort on the GDCs in 10CFR50 Appendix A?<br>• What are NRC's plans and schedule for risk-informing the GDCs? |
| **Other Advanced Non-LWR Activities** | Paragraph 1: In addition to providing design criteria related to safety considerations, the staff is contemplating design considerations related to security requirements. This information is forthcoming and will be issued for comment separately. | Design considerations and associated regulatory requirements related to security are currently addressed outside of 10 CFR 50 Appendix A. This structure should be maintained, and design considerations related to security should not be incorporated into the advanced reactor design criteria, unless similar updates are reflected in Appendix A. |
| **Topics for Open for Comment** | Should the current regulations that an applicant must address be incorporated into the ARDC? If so, which ones? | Appendix A of 10 CFR 50 does not currently contain a list of the current regulations that an applicant must address. Regulations that are not currently reflected in 10 CFR 50 Appendix A should similarly be excluded from the ARDC.<br><br>It is noted that this same question was raised during an NRC public meeting held on January 21, 2015 (meeting summary available at ML15044A081). The NRC and DOE responded that this was beyond the scope of this initiative but could be a future endeavor.<br><br>DOE would continue to agree that this should be considered as a future endeavor, but it is outside the scope established for this initiative, and should not be incorporated into the current regulatory guidance development effort for a number of reasons, including:<br>a) The ARDC would then be inconsistent with the structure and content of the GDC, and outside the scope of both this initiative and the "adaptation" language reflected in the introductory language in Appendix A.<br>b) Rulemaking would likely be needed to align the GDC and ARDC.<br>c) Incorporation of current regulations into the ARDC would require significant effort and likely a deferral of the planned schedule for Regulatory Guide issuance, extending the current state of regulatory uncertainty for advanced reactor stakeholders regarding design criteria guidance. |
| | Are the ARDC generally applicable to the different types of non-LWRs being developed by different companies? | As discussed in the January 21, 2015 NRC public meeting (summary available at ML15044A081), the DOE confirmed that the proposed ARDC's were generally applicable to six advanced reactor designs, Sodium-cooled Fast Reactors (SFRs), Lead Fast Reactors (LFRs), Gas-cooled Fast Reactors (GFRs), modular High Temperature Gas-cooled Reactors (HTGRs), Fluoride High Temperature Reactors (FHRs), and Molten Salt Reactors (MSRs), based on the information available at that time. |
| | | The Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" (ML082750370) summarizes its expectation that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.<br><br>NRC's proposed content for ARDC 17 appears to be in conflict with this Commission expectation since the offsite power requirements typically associated with plants that rely on active safety systems are retained. This apparently conflicting ARDC content may discourage advanced reactor designers from pursuing the Commission's policy expectations regarding reduced reliance on active safety systems, since they'd then need to pursue a departure or exemption from this ARDC during the license application review |

| DOE Laboratory Team Comments on NRC Invitation and Instructions | | |
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| **Section** | **NRC Language** | **Team Comments** |
| | | process. |
| | Is the approach to "functional containment" appropriately addressed in the proposed criteria? | The approach to "functional containment" is not appropriately addressed, since NRC did not incorporate the concept into ARDC 16. This creates a disconnect between this overarching ARDC and the more technology specific mHTGR-DC 16 (i.e., mHTGR-DC 16 will not logically derive from ARDC 16).<br><br>The ARDC language proposed by DOE in the December 2014 report (ML14353A246 and ML14353A248) was intended to be sufficiently general to allow the other reactor technology versions of Criterion 16 to be the same as the ARDC with no further technology specific modifications. The language was chosen to encompass either a traditional containment structure that surrounds the reactor and its coolant system or a technology specific functional containment such as that used by the modular HTGR. It would be incumbent on the designer to demonstrate the suitability of their design for their reactor type and to demonstrate compliance with regulatory requirements for offsite dose.<br><br>The more narrow ARDC language suggested by NRC is not consistent with the Commission SRM issued in response to SECY-93-0092, the SRM issued in response to SECY-03-0047, and SECY-05-006. The proposed NRC language removes flexibility and may dissuade future designers from pursuing technology specific functional containment concepts. The NRC should reflect the "functional containment" concept proposed by DOE in ARDC 16, and require advanced reactor applicants to justify their designs. |
| **Role of GDC in Regulatory Framework** | *The NRC had anticipated exercising the NUREG-1860 process for the Next Generation Nuclear Plant (NGNP) project. However, when NGNP was terminated by DOE, the NRC no longer had a viable demonstration project.* | The DOE did not terminate the NGNP project. In a letter from Energy Secretary Chu to Congress (October 17, 2011), the Secretary confirmed that the Department would not proceed with design activities at that time. However, the letter also confirmed a continued focus on high temperature reactor research and development activities, and interactions with the Nuclear Regulatory Commission to develop a licensing framework. |
| **NRC Policy on Advanced Reactors** | Sentence 2: *From the NRC staff's regulatory perspective, the characteristics of an "advanced reactor" has evolved over time, and this evolution is expected to continue.* | "has" should be "have". |
| **Role of GDC for Advanced Non-LWRs** | Last Paragraph: *The NRC had anticipated exercising the NUREG-1860 process for the Next Generation Nuclear Plant (NGNP) project.* | NUREG-1860 was written by the NRC Research staff. During discussions with NRO staff during NGNP reviews, the status of NUREG-1860 was never made clear and the staff never indicated official intent to use NUREG-1860. This discussion should make reference to NUREG-2150, which should be playing a significant role in regulation of advanced non-LWRs as well as other advanced reactors. |
| **Topics for Open for Comment** | *NRC is seeking also input regarding: use of "functional containment" for mHTGR-DC, and use of "specified acceptable radionuclide release design limits" (SARRDLs) in the mHTGR-DC in place of specified acceptable fuel design limits (SAFDLs)* | SARRDL refers to a "specified acceptable **core** radionuclide release design limit." The quoted NRC text did not include the word "core" and should be revised accordingly. Core radionuclide release is a parameter that is directly related to fuel performance and can be directly monitored during plant operation. The limit does not include radionuclide release from other points such as the reactor helium pressure boundary or the reactor building. |
| | | Although NRC is requesting comment on whether these approaches are appropriately addressed in these design criteria, broader comments will likely be received regarding the acceptability of the overall design approach for these advanced reactors, especially from members of the intervener community. |

| DOE Laboratory Team Comments on Cross-Cutting Topics | | |
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| **Cross-cutting Topic** | **NRC Language** | **Team Comments** |
| Application of "Important to Safety" (ITS) | Proposed ARDC 3 suggests that NRC considers ITS to be applicable to SSCs in capacities other than denoting safety related equipment. | On page 2 of the introductory material, NRC discusses Important-to-Safety (ITS), but it is not clear what the discussion implies. It is also noted that the proposed ARDC-3 language indicates that there is a difference between SSCs that are important to safety and SSCs that are safety related (SR). Note that ITS is used throughout the GDC. SR is not used anywhere in the GDC.<br><br>The DOE December 2014 report (ML14353A246 and ML14353A248) provided a definition of the term "important to safety" to clarify and confirm its use, based on industry's understanding of its intended meaning within the context of 10 CFR 50 Appendix A. This term was further addressed in DOE's response to NRC Question 40 (ML15204A579). The need for this clarification was also discussed in an NRC public meeting on January 21, 2015 (meeting summary available at ML15044A081), with DOE indicating that the ARDC's could be heavily impacted if NRC's understanding of the use of the term within Appendix A is different from that provided in the clarifying definition.<br><br>NRC should either confirm that "important to safety" means "safety related" within the context of Appendix A, or explain the difference between the two terms and provide a regulatory basis. |
| Need to review and provide feedback on proposed definitions. | N/A | The DOE report provided a series of Definitions (report Section 3.1) that are intended to confirm a common understanding of the use of certain terms within the context of 10 CFR 50 Appendix A, and to provide added clarity regarding the use of selected terms unique to DOE's proposed non-LWRs design criteria. Understanding the Definitions in the DOE report is essential to a common understanding of the ARDC requirements, and of the requirements associated with technology-specific examples of design criteria derived from the ARDCs.<br><br>These definitions were reviewed with the NRC staff in a public meeting on January 21, 2015 (meeting summary available at ML15044A081), but were not reflected in the material provided for public comment. They should be addressed and confirmed in the pending Regulatory Guide. |
| Use of "as appropriate" or "as necessary" | Wording was removed in ARDC 13 and 55 | Some SFR team members believe this wording change could pose significant impact. |

| DOE Laboratory Team Comments on NRC ARDC<br>I. Overall Requirements | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 1 | Same as GDC<br><br>*Quality standards and records.*<br>Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit. | No comments. |
| 2 | Same as GDC<br><br>*Design bases for protection against natural phenomena.*<br>Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed. | No comments. |
| 3 | *Fire protection.*<br>Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations ~~such as the containment and control room~~ with safety-related equipment or structures, systems, and ~~or~~ components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided | The term “safety related equipment” in the second sentence of the ARDC appears to be redundant to the term “structures, systems, and components important to safety” and potentially changes the applicability of the DC as stated in the first sentence. As written, this text makes a distinction between “safety related equipment” and “structures, systems, and components important to safety” that is made no place else in the design criteria. The words, “safety related equipment or” in the second sentence of the proposed ARDC should be removed to eliminate the confusion. |

| DOE Laboratory Team Comments on NRC ARDC<br>I. Overall Requirements | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.<br><br>Rationale<br><br>The phrase containing examples where noncombustible and heat resistant materials must be used has been broadened to apply to all advanced reactor designs. | The ITS ambiguity in existing NRC regulations and guidance was identified during the DOE ARDC development. The understood meaning of ITS within the context of 10 CFR 50 Appendix A was defined on page 7 of the DOE report (ML14353A246 and ML14353A248) and was further addressed in DOE's response to NRC Question 40 (ML15204A579). The need for this clarification was also discussed in an NRC public meeting on January 21, 2015 (meeting summary ML15044A081), with DOE indicating that the ARDCs could be heavily impacted if NRC's understanding of the use of the term within Appendix A is different from that provided in the clarifying definition. The NRC's use of ITS in ARDC 3 seems to deviate from that understanding.<br><br>Consistency with the GDC is needed with regard to the use of "important to safety" in the ARDC. "Important to safety" is broadly used throughout the GDC; "safety related" is not used anywhere in the GDC. NRC should either confirm that "important to safety" means "safety related" within the context of Appendix A, or explain the difference between the two terms and provide a regulatory basis.<br><br>It is also recommended that the NRC's rational should be reworded as follows: "The phrase containing examples where noncombustible and heat resistant materials must be used has been broadened to apply to SSCs important to safety throughout the unit." |
| 4 | *Environmental and dynamic effects design bases.*<br>Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents~~, including loss of coolant accidents~~. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.<br><br>Rationale | It is not clear why the last sentence in the rationale for ARDC 4 has been inserted by the staff. For low pressure systems, it should be easy for designers and applicants to demonstrate that pipe whip phenomena do not apply, as implied in the last sentence of the criterion, which has not been modified for the ARDC. This last sentence in the rationale does not directly tie to any revision of this criterion, and it should be deleted. |

| DOE Laboratory Team Comments on NRC ARDC<br>I. Overall Requirements | | |
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| | This change removes the LWR emphasis on loss of cooling accidents (LOCAs) that may not apply to some designs. For example, helium is not needed in a mHTGR to remove heat from the core during postulated accidents and does not have the same importance as water does to LWR designs to assure that fuel integrity is maintained. Therefore, a specific reference to "loss of coolant accidents" is not applicable to all designs. LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design. Reference to pipe whip may not be applicable to designs that operate at low pressure. | |
| 5 | Same as GDC<br><br>*Sharing of structures, systems, and components.*<br>Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>II. Multiple Barriers | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 10 | Same as GDC<br><br>*Reactor design.*<br>The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. | No comments. |
| 11 | *Reactor inherent protection.*<br>The reactor core and associated ~~coolant~~ systems that contribute to reactivity feedback shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.<br><br>Rationale<br><br>The wording has been changed to broaden the applicability from "coolant systems" to additional factors (including structures or other fluids) that may contribute to reactivity feedback. These systems are to be designed to compensate for rapid reactivity increase. | No comments. |
| 12 | *Suppression of reactor power oscillations.*<br>The reactor core and associated structures, coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.<br><br>Rationale<br><br>The word "structures" was added because items such as reflectors, which could be considered either outside or not part of the reactor core, may affect susceptibility of the core to power oscillations. | No comments. |
| 13 | *Instrumentation and control.*<br>Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions ~~as appropriate~~ to assure adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core**,** the reactor coolant ~~pressure~~ boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>II. Multiple Barriers | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | systems within prescribed operating ranges.<br><br>Rationale<br><br>"As appropriate" was removed to provide specificity to the criterion. "Reactor coolant pressure boundary" has been relabeled as "reactor coolant boundary" to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. | |
| 14 | *Reactor coolant ~~pressure~~ boundary.*<br>The reactor coolant ~~pressure~~ boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "reactor coolant boundary" to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non- LWRs that operate at either low or high pressure. | No comments. |
| 15 | *Reactor coolant system design.*<br>The reactor coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor coolant ~~pressure~~ boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.<br><br>Rationale<br><br>Reactor coolant pressure boundary has been relabeled as "reactor coolant boundary" to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non- LWRs that operate at either low or high pressure. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>II. Multiple Barriers | | |
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| 16 | Same as GDC<br><br>*Containment design.*<br>Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.<br><br>Rationale<br><br>For non-LWR technologies other than SFRs and mHTGRs, designers should use the current GDC to develop applicable principal design criteria. | DOE-proposed ARDC 16 language, in conjunction with the definition of "functional containment" as provided in the DOE report, is sufficient to address radiological containment for a wide variety of advanced reactors without incurring technology-specific modifications. DOE's proposed criterion text was selected to encompass both traditional containment structures (and associated systems) and multi-barrier functional containment approaches and maintains consistency with Commission SRMs issued in response to SECY-93-0092, in response to SECY-03-0047, and with SECY-05-0006. For example, the SRM on SECY-03-0047 directed the staff to develop containment performance requirements and criteria working closely with industry and other stakeholders. ARDC 16 should be revised to reflect the DOE proposal. |
| 17 | *Electric power systems.*<br>An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant ~~pressure~~ boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.<br><br>The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.<br><br>Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the | NRC's proposed ARDC 17 retains electric power requirements that may have no safety application in certain advanced reactor designs. As such, this criterion does not apply to many advanced reactor systems.<br><br>Criterion Paragraph 1 requires that, "(2) the core is cooled and containment integrity and other vital functions are maintained…" By specifically encompassing core cooling and containment integrity as well as other vital functions that rely on electric power, the text becomes too broad for the ARDC. Because ARDCs allow for flexible designs, the DOE proposed ARDC 17 text should be used to address the requirement. This comment also applies to the proposed SFR and modular HTGR design criteria.<br><br>Components listed in the criterion (such as batteries and onsite electric distribution systems) are very specific and could impede innovations that mitigate the need for such safety systems if retained as a design requirement. This adherence to LWR electric power system requirements appears to be in conflict the Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" (ML082750370) which summarizes its expectation that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.<br><br>Advanced reactor design criterion requirements should focus on ensuring that sufficient time and capability is provided following a postulated accident to assure core cooling, containment integrity, and other vital safety functions are maintained. ARDC 17 (as currently proposed by NRC) neither recognizes nor |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | reactor coolant ~~pressure~~ boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a postulated ~~loss-of- coolant~~ accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.<br><br>Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.<br><br>Rationale<br><br>The requirements for offsite power are being retained for defense-in-depth considerations. This position was reinforced by a letter from the NRC to Dale Atkinson, Chief Operating Officer, NuScale Power, September 15, 2015 (ML15222A323). At the September 24, 2015 meeting of the Advisory Committee for Reactor Safeguards subcommittee on advanced reactor designs, this subject came up again and the subcommittee was supportive of keeping offsite power requirements in GDC 17 for the NuScale design.<br><br>LWR emphasis on LOCAs may not apply to non-LWR designs. For example, helium is not needed in an HTGR to remove heat from the core during postulated accidents and does not have the same importance as water does to LWR designs to assure that fuel integrity is maintained. LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design.<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. | credits the unique safety features that will be associated with emerging advanced reactor designs.<br><br>The reference to the ACRS September 24 subcommittee meeting in the rationale should be removed since the subcommittee was addressing the NuScale DSRS and requested the staff to remain flexible in its wording so as to accommodate the possibility that NuScale may not need AC power to accomplish system safety functions after review of their design.<br><br>NRC should adopt the DOE-proposed ARDC 17 language. This language focuses on requiring electric power as necessary to ensure safety functions are performed when required. It also provides relief from off-site AC power source and on-site safety-grade AC power source requirements for those designs that can successfully accomplish safety functions, assuming a single failure, without needing AC electric power. This approach is flexible, focused, and aligned with guidance contained in the Commission’s 2008 “Policy Statement on the Regulation of Advanced Reactors” (ML082750370).<br><br>Please note that modifications to ARDC 17 to address the needs of advanced non-LWRs also necessitate changes to text contained in ARDCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 18 | Same as GDC<br><br>*Inspection and testing of electric power systems.*<br>Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to | No comments. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.<br><br>Rationale<br><br>GDC 18 is a design- independent companion criterion to GDC 17. | |
| 19 | *Control room.*<br>A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions~~including loss-of-coolant accidents~~. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) ~~whole body, or its equivalent to any part of the body~~, (TEDE) as defined in § 50.2 for the duration of the accident.<br><br>Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions.<br><br>Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.<br><br>~~Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January~~ | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>II. Multiple Barriers | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | ~~10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.~~<br><br>Rationale<br><br>The criterion was updated to remove specific emphasis on LOCA, which may be not appropriate for advanced designs such as the mHTGR.<br><br>Reference to “whole body, or its equivalent to any part of the body” has been updated to the current TEDE standard as defined in § 50.2.<br><br>Control room habitability requirement beyond that associated with radiation protection has been added to address concern that non-radionuclide accidents may also affect control room access and occupancy.<br><br>The last paragraph of the GDC has been eliminated for the ARDC because it is not applicable to future applicants. | |

| DOE Laboratory Team Comments on NRC ARDC<br>III. Reactivity Control | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 20 | Same as GDC<br><br>*Protection system functions.*<br>The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.<br><br>Rationale<br><br>For non-LWR technologies other than mHTGRs designers should use the current GDC to develop applicable principal design criteria. | No comments. |
| 21 | Same as GDC<br><br>*Protection system reliability and testability.*<br>The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. | No comments. |
| 22 | Same as GDC<br><br>*Protection system independence.*<br>The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>III. Reactivity Control | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 23 | Same as GDC<br><br>*Protection system failure modes.*<br>The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.<br><br>Rationale<br><br>For non-LWR technologies other than SFRs, designers should use the current GDC to develop applicable principal design criteria. | No comments. |
| 24 | Same as GDC<br><br>*Separation of protection and control systems.*<br>The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired. | No comments. |
| 25 | *Protection system requirements for reactivity control malfunctions.*<br>The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded during any anticipated operational occurrence resulting from a ~~for any~~ single malfunction of the reactivity control systems. ~~, such as accidental withdrawal (not ejection or dropout) of control rods~~<br><br>Rationale<br><br>Text has been added to clarify that the protection system is designed to protect the SAFDLs for AOOs in combination with a single failure; the protection system does not have to protect the SAFDLs during a postulated accident in combination with a single failure. The example was deleted to make ARDC technology neutral. | No comments. |

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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 26 | *Reactivity control system redundancy and capability.*<br>At least two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes ~~(including xenon burnout)~~ to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.<br><br>Rationale<br><br>"At least" was added to set a minimum number of independent reactivity control systems; it does not preclude more than two systems.<br><br>The parenthetical phrase "including xenon burnout" has been deleted as it is already addressed by the statement "…rate of reactivity changes resulting from planned, normal power changes." In other words, the second reactivity control system must control the reactivity changes relevant to the specific design for normal plant power changes. This deletion makes the ARDC more technology neutral. For example, xenon burnout does not apply to fast reactor designs.<br><br>"Cold conditions" remains but will have to be defined by a principal design criteria for the specific design. | Last Sentence of Rationale: This sentence should be deleted since it does not relate to a change in the design criteria language. The requirement for subcriticality under cold conditions from the original GDC 26 was not changed in the ARDC or in the SFR and modular HTGR criteria. The sentence was taken by the NRC staff from the rationale for SFR DC 26 in the DOE report. "Cold conditions" are typically defined in facility Technical Specifications (as was the case at Fort St. Vrain) or in safety analyses, but not in principal design criteria. |
| 27 | *Combined reactivity control systems capability.*<br>The reactivity control systems shall be designed to have a combined capability~~, in conjunction with poison addition by the emergency core cooling system,~~ of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.<br><br>Rationale<br><br>None of the advanced non- LWR designs evaluated in the review utilized poison addition via an ECCS. | We agree with the NRC rationale for ARDC-27. However, the NRC staff rationale presented for ARDC-27 is inconsistent with the NRC staff position to retain ARDC-35. |

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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | In addition, ARDC 34, *Residual heat removal*, combines the ECCS requirements in GDC 35 into ARDC 34, because none of the advanced non-LWR designs evaluated utilized an ECCS. Advanced non-LWR designs that do use poison addition or an ECCS will have to look to GDC 27 and GDC 35 for guidance. | |
| 28 | *Reactivity limits.*<br>The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant ~~pressure~~ boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor ~~pressure~~ vessel internals to impair significantly the capability to cool the core. ~~These postulated reactivity accidents shall include consideration of [rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition].~~<br><br>Rationale<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.<br><br>The word “pressure” was deleted when referring to the reactor vessel as some designs may not be pressurized (SFR for example).<br><br>The list of “postulated reactivity accidents” has been deleted to make the ARDC technology neutral. Each design will have to determine its postulated reactivity accidents based on the specific design and associated risk evaluation. | No comments. |
| 29 | Same as GDC<br><br>*Protection against anticipated operational occurrences.*<br>The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 30 | *Quality of reactor coolant ~~pressure~~ boundary.*<br>Components which are part of the reactor coolant ~~pressure~~ boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.<br><br>Rationale<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. | No comments. |
| 31 | *Fracture prevention of reactor coolant ~~pressure~~ boundary.*<br>The reactor coolant ~~pressure~~ boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.<br><br>Rationale<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. | No comments. |
| 32 | *Inspection of reactor coolant ~~pressure~~ boundary.*<br>Components which are part of the reactor coolant ~~pressure~~ boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor ~~pressure~~ vessel. | No comments. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | Rationale<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.<br><br>The staff modified the LWR GDC by replacing the term “reactor pressure vessel” with “reactor vessel”, which staff believes is a more generically applicable term. | |
| 33 | *Reactor coolant inventory maintenance~~makeup~~.*<br>A system to maintain ~~supply~~ reactor coolant inventory ~~makeup~~ for protection against small breaks in the reactor coolant ~~pressure~~ boundary shall be provided as necessary. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant inventory loss due to leakage from the reactor coolant ~~pressure~~ boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain coolant inventory during normal reactor operation.<br><br>Rationale<br><br>Retitled with “inventory maintenance” to provide more flexibility regarding advanced reactor designs.<br><br>The term “...shall be provided as necessary to assure...” has been modified to recognize the inventory control system may be unnecessary for some designs to maintain safety functions that assure fuel design limits are not exceeded.<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to | Recognizing that the inventory control system may be unnecessary for some designs to maintain safety functions that assure fuel design limits are not exceeded, criterion sentences 1 and 2 should delete the words, “The system safety function shall be…”, and be combined to read: “A system to maintain reactor coolant inventory for protection against small breaks in the reactor coolant boundary shall be provided as necessary to assure that the …”.<br><br>The comment brings NRC ARDC 33 text back into alignment with DOE-proposed wording. For low pressure advanced reactor primary systems, small breaks are not credible sources of significant coolant inventory loss in a short time period. The mHTGR design does not require helium makeup during postulated accidents. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | non-LWRs that operate at either low or high pressure. Maintained the words "system safety function" of GDC 33 as reactor coolant inventory maintenance may be necessary in some designs to support residual heat removal which is a safety function. If not required for maintaining residual heat removal capability the qualifier "as necessary" in the first sentence would apply. For example, if all small breaks or leaks would result in reactor coolant inventory levels such that residual heat removal function would still be performed, and the fuel design limits met, no safety function would be associated with the inventory maintenance system. | |
| 34 | *Residual heat removal.*<br>A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the ~~The~~ system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant ~~pressure~~ boundary are not exceeded.<br><br>During postulated accidents, the system safety function shall provide continuous effective core cooling and to assure that the design conditions of the reactor coolant boundary are not exceeded.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>ARDC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35.<br><br>"Ultimate heat sink" has been added to clarify that if ARDC 44 is deemed not applicable to the design, the RHR system is then required to provide the heat removal path to the ultimate heat sink.<br><br>Reactor coolant pressure boundary has been relabeled as "reactor coolant boundary" to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating | DOE has previously reviewed the content and structure of the existing General Design Criteria included in Appendix A of 10 CFR 50, and has determined that various criteria (such as GDC's 4, 14, and 31) address the aspects of reactor coolant boundary integrity during postulated accidents. The underlying bases established in those criteria are retained in the adaptations previously proposed in the submitted DOE report for the corresponding ARDC's. In addition, exceeding reactor coolant boundary design conditions can be the initiating event for some postulated accidents which is not consistent with the NRC proposed words in the second paragraph of the ARDC. Therefore, the second paragraph of ARDC 34 should be revised to "During postulated accidents, the system safety function shall provide continuous effective core cooling."<br><br>It is also noted that a revised proposal for ARDC 34 content that provides further clarity was submitted by DOE on September 15, 2015 (ML15272A096). ARDC 34 should be updated to reflect that revised format and content. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.<br><br>Text of first paragraph has been amended and the second paragraph added to clarify requirements that are applicable following normal operation including AOOs, and during postulated accidents following the precedent of NUREG-1368, "Pre-application SER for PRISM LMR."<br><br>The last phrase was added to the second paragraph to assure that residual heat removal capability is sufficient to maintain the integrity of the reactor coolant boundary during postulated accidents. Maintaining the reactor coolant boundary is wording not currently in GDC 35 as the limiting postulated accident is a LOCA where primary coolant integrity is assumed lost. In advanced designs other accidents may be more limiting than a LOCA and hence the residual heat removal capability should be designed to ensure the reactor coolant boundary integrity is maintained.<br><br>The third paragraph addresses RHR system redundancy. ARDC 17 requires reliable power systems for SSCs performing vital safety functions and must be of adequate capacity and capability to operate during postulated accidents. There may be various combinations of power supply employed to address power reliability. | |
| 35 | *Emergency core cooling.*<br>If the system as described in ARDC 34 does not provide continuous effective core cooling during postulated accidents and does not assure that the design conditions of the reactor coolant boundary are preserved; then a ~~A~~ system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant such that continuous effective core cooling is maintained. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale | DOE did not propose ARDC content, but instead proposed that; "If a separate ECCS system is required for an advanced reactor, the PDC process for that reactor must look directly to GDC 35 for guidance." This does not suggest an advanced reactor design equivalent to GDC 35 is needed given that the proposed ARDC 35 does not support any known non-LWR technology. If a system described in ARDC 34 does not provide continuous effective core cooling during postulated accidents (which is required), the system is improperly designed and subject to revision to meet the ARDC 34 requirement. A "contingency" system should not be assumed as an arbitrary design option. Criterion 35 is obsolete and not applicable to advanced non-LWRs.<br><br>Also note that the NRC staff rationale presented in ARDC-27 is inconsistent with the NRC staff position to retain ARDC-35. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | In most advanced reactor designs, residual heat removal is addressed by ARDC 34. If the design is such that ARDC 34 is not adequate to ensure residual heat removal under normal operations and postulated accidents then additional system(s) are required and would be addressed by this ARDC 35 to ensure continuous effective core cooling. | |
| 36 | *Inspection of ~~emergency core~~ ~~cooling~~ residual heat removal system.*<br>The ~~emergency core cooling~~ ~~system~~ residual heat removal shall be designed to permit appropriate periodic inspection of important components, ~~such as spray rings in the reactor pressure vessel, water injection nozzles, and piping,~~ to assure the integrity and capability of the system.<br><br>Rationale<br><br>Title has been renamed and GDC revised to provide for inspection of the residual heat removal systems as required for ARDC 34.<br><br>If emergency core cooling is required (ARDC 35) then it shall be designed to permit appropriate periodic inspection of important comments.<br><br>The example list has been deleted because it applies to LWR designs and each specific design will have different important components associated with residual heat removal. This revision allows for a technology neutral ARDC.<br><br>Review of the proposed DOE SFR and HTGR DCs found that only SFR provided specific examples of important components but were generic in nature and did not add any significant additional guidance. | No comments. |
| 37 | *Testing of residual heat removal ~~emergency core~~ ~~cooling~~ system.*<br>The residual heat removal ~~emergency core cooling~~ system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of associated systems and interfaces with an ultimate heat sink including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system. | Some advanced reactor residual heat removal systems do not operate at pressure. The first sentence of the criterion should therefore be changed from "…periodic pressure and functional testing…" to "…periodic functional testing". Functional testing can include pressure testing as appropriate and the required pressure capability specified in test acceptance criteria.<br><br>Certain mHTGR reactor cavity cooling systems are open at each end, thus eliminating the need for leaktight integrity. Item (1) words "…structural and leaktight integrity…" should therefore be changed to "…structural integrity". Structural integrity can include leaktightness as appropriate and the degree of leaktightness specified in test acceptance criteria. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | Rationale<br><br>GDC 37 system has been renamed and revised to provide for testing of the residual heat removal system of ARDC 34.<br><br>If emergency core cooling is required (ARDC 35) then it shall be designed to permit appropriate periodic pressure and functional testing of its components.<br><br>A specific requirement for pressure and leaktight testing was retained in the ARDC as future advance designs may employ pressure retaining RHR designs. If the applicable system in the advanced design is not pressure retaining, then "periodic pressure testing" and "leaktight integrity" could be removed in the specific design criteria.<br><br>"Active" has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature.<br><br>Reference to operation of applicable portions of the protection system, cooling water system, and power transfers is considered part of the more general "associated systems." Together with the ultimate heat sink, they are part of the operability testing of the system as a whole. | The last sentence of the criterion states "…including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system…" which should be deleted. The NRC criterion rationale notes that operation of applicable portions of the protection system, cooling water system, and power transfers is considered part of the more general "associated systems" and together with the ultimate heat sink, they are part of the operability testing of the system as a whole. This suggests the criterion statement was intended for deletion. |
| 38 | *Containment heat removal.*<br>A system to remove heat from the reactor containment shall be provided as necessary ~~The system safety function shall be~~ to maintain ~~reduce rapidly, consistent with the functioning of other associated systems,~~ the containment pressure and temperature within acceptable limits following ~~following any loss-of-coolant~~ postulated accidents. ~~and maintain them at acceptably low levels.~~<br><br>Suitable redundancy in components and features, including electric power systems , and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. | The basis for adding "…including electric power systems" to criteria 38 and 41 is unclear and not addressed in the rationale. Electric power systems are included under components and features, thereby making the new text unnecessary. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | Rationale<br><br>"…as necessary…" is meant to condition ARDC requiring heat removal for conventional containments which are found to require heat removal measures.<br><br>LOCA reference has been removed to provide for any postulated accident that might affect the containment structure.<br><br>Containment structure safety system redundancy is addressed in second paragraph. | |
| 39 | *Inspection of containment heat removal system*.<br>The containment heat removal system shall be designed to permit appropriate periodic inspection of important components~~, such as the torus, sumps, spray nozzles, and piping~~ to assure the integrity and capability of the system.<br><br>Rationale<br><br>Examples were deleted to make the ARDC technology neutral. | No comments. |
| 40 | *Testing of containment heat removal system.*<br>The containment heat removal system shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the structural ~~and leak-tight~~ integrity of its components, (2) the operability and performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system, including operation of associated systems.<br><br>Rationale<br><br>Specific mention of "pressure" testing has been removed yet remains a potential requirement should it be necessary as a component of "…appropriate periodic functional testing..." of cooling systems. | The words under Criterion Item (3), "…including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system," are addressed "...under operation of associated systems" and should be deleted. The last paragraph of the NRC rationale suggests the statement was intended for deletion. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | "Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability.<br><br>Reference to operation of applicable portions of the protection system, cooling water systems, and power transfers is considered part of the more general "associated systems" for operability testing of the system as a whole. | |
| 41 | *Containment atmosphere cleanup.*<br>Systems to control fission products ~~hydrogen, oxygen~~ and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of ~~hydrogen or~~ ~~oxygen and~~ other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.<br><br>Each system shall have suitable redundancy in components and features, including electric power systems, and suitable interconnections, leak detection, isolation, and containment capabilities to assure ~~that~~ that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>Advanced reactors offer potential for reaction product generation that is different from that associated with clad metal-water interactions. Therefore, the terms "hydrogen" and "oxygen" are removed while "other substances" is retained to allow for exceptions. | The basis for adding the words "…including electric power systems" to the criterion is unclear and not addressed in the rationale. Since electric power systems are included under components and features, this new text is redundant and should be deleted. |
| 42 | Same as GDC<br><br>*Inspection of containment atmosphere cleanup systems.*<br>The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 43 | *Testing of containment atmosphere cleanup systems.* The containment atmosphere cleanup systems shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the structural ~~and leak-tight~~ integrity of its components, (2) the operability and performance of the ~~active~~ system components, ~~of the systems such as fans, filters, dampers, pumps, and valves~~ and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and including the operation of associated systems.<br><br>Rationale<br><br>"Active" has been deleted in item (2) as appropriate operability and performance testing of system components is required regardless of active or passive nature, as are cited examples of active system components.<br><br>Examples of active systems under item (2) have been deleted both to conform to similar wording in ARDC 37 and 40 and ensure passive as well as active system components are considered.<br><br>Specific mention of "pressure" testing has been removed yet remains a potential requirement should it be necessary as a component of "…appropriate periodic functional testing..." of cooling systems.<br><br>"Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability. | The word "including" is misplaced in the last phrase of the criterion relative to GDC 43. It should follow the phrase, "brings the systems into operation." |
| 44 | *Structural and equipment cooling ~~Cooling water~~.*<br>In addition to the heat rejection capability of the residual heat removal system and the emergency core cooling system (if provided), ~~A~~ systems to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided, as ~~n~~ecessary. ~~The system safety function shall be~~ to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be | No comments. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor design system differences to include safety-related cooling requirements for SSCs, if applicable; this ARDC does not address the residual heat removal system required under ARDC 34 or the emergency core cooling system required under ARDC 35 (if required). | |
| 45 | *Inspection of structural and equipment cooling ~~water~~* systems.<br>The ~~cooling water~~ structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the systems.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs. | No comments. |
| 46 | *Testing of structural and equipment cooling ~~. water~~ systems.*<br>The structural and equipment cooling ~~water~~ systems shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the structural ~~and leaktight~~ integrity of their ~~its~~ components, (2) the operability and the performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring~~s~~ the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems. ~~and for loss-of-coolant accidents, including operation of~~ and applicable portions of the protection system and the transfer between normal and emergency power sources.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>IV. Fluid Systems | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | Specific mention of “pressure” testing has been removed yet remains a potential requirement should it be necessary as a component of “...appropriate periodic functional testing...” of cooling systems.<br><br>“Leaktight” integrity would be demonstrated through appropriate functional testing of system performance and operability.<br><br>“Active” has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature.<br><br>LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs. | |

| DOE Laboratory Team Comments on NRC ARDC<br>V. Reactor Containment | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 50 | *Containment design basis.*<br>The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. ~~any loss-of-coolant accident~~. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as fission products, potential spray or aerosol formation, and potential exothermic chemical reactions ~~energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning~~, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.<br><br>Rationale<br><br>ARDC-50 specifically addresses a containment structure in the opening sentence and ARDCs 51-57 support the containment structure's design basis.<br>Therefore, ARDC 51 – 57 are modified by adding the word "structure" to highlight the containment structure-specific criteria.<br><br>The phrase "loss of coolant accident" is LWR-specific because this is understood to be the limiting containment structure accident for an LWR design. It is replaced by the phrase "postulated accident" to allow for consideration of the design-specific containment structure limiting accident for advanced non- LWR designs.<br><br>The example at the end of subpart 1 of the ARDC is LWR-specific and therefore deleted. | The rationale notes the deletion of the LWR-specific example in subpart 1 of the ARDC. However, the rationale does not address the selection of the replacement examples selected for inclusion in ARDC-50.<br><br>For clarity and consistency, a rationale should be provided for all text changes or the subpart 1 example list should simply be eliminated as supported by the current rationale text. |
| 51 | *Fracture prevention of containment pressure boundary.*<br>The ~~reactor containment~~ boundary of the reactor containment structure shall be designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ~~ferritic~~ materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration | In the rationale, the second half of the last sentence states "that this criterion is generically applicable to all non-LWR designs." Given that Criterion 51 does not apply to the mHTGR design, the phrase should be restated "that this criterion is more broadly applicable to non-LWR designs." |

| DOE Laboratory Team Comments on NRC ARDC<br>V. Reactor Containment | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining<br>(1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC- 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure.<br>Therefore, the word “structure” is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct.<br><br>The term “ferritic” was removed in order to not limit the scope of the criterion to ferritic materials. With this revision, the staff believes that this criterion is generically applicable to all non-LWR designs. | |
| 52 | *Capability for containment leakage rate testing*.<br>The reactor containment structure and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word “structure” is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct. | No comments. |
| 53 | *Provisions for containment testing and inspection.*<br>The reactor containment structure shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows. | No comments. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word "structure" is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct. | |
| 54 | *Piping systems penetrating containment.*<br>Piping systems penetrating the ~~primary~~ reactor containment structure shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. ~~Such P~~piping systems shall be designed with the ~~a~~ capability to verify by testing ~~periodically~~ the ~~operability of the~~ operational readiness of any isolation valves and associated apparatus periodically, ~~and to~~ ~~determine if~~ and to confirm that valve leakage is within acceptable limits.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word "structure" is added to each of these ARDCs to clearly convey the understanding that this ARDC only applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct. The adjustment to the last sentence enhances the clarity of the sentence with respect to the latest terminology used for valve periodic verification and operational readiness.<br><br>The ASME Operation and Maintenance of Nuclear Power Plants, Division 1: OM Code: Section IST (ASME OM Code) defines operational readiness as the ability of a component to perform its specified functions. The ASME OM Code is incorporated by reference in the NRC regulations in 10 CFR 50.55a, including the definition of operational readiness for pumps, valves, and dynamic restraints. | The last sentence of the criterion should be left unchanged. Although it is recognized that the proposed revisions are intended to align the language with the associated ASME Standard, that Standard is already incorporated by reference, thus confirming the technical expectation. Changing the criterion text in this pending Regulatory Guide could create future inconsistencies between the criterion and the ASME Standard, if the Standard is updated and endorsed by the NRC in the future.<br><br>In addition, changing the criterion text does not appear to provide technology-neutral language that is distinctly necessary to differentiate the design criterion for advanced non-LWR reactor designs from the current design criterion for LWR designs as specified in GDC-54. |
| 55 | *Reactor coolant ~~pressure~~ boundary penetrating containment.*<br>Each line that is part of the reactor coolant ~~pressure~~ boundary and that penetrates the ~~primary~~ reactor containment structure shall be provided with containment isolation valves as follows, unless it can be | (JK/RB) No rationale is provided for the deletion of "as necessary" from the text of this criterion. This deletion does not appear to be needed and could be viewed by some as a reduction in current levels of flexibility. The existing GDC "as necessary" text should be retained. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:<br>(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or<br>(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or<br>(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or<br>(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.<br><br>Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.<br><br>Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided ~~as necessary~~ to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word “structure” is added to each of these ARDCs to clearly convey the understanding that this ARDC only applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct.<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that | |

| DOE Laboratory Team Comments on NRC ARDC<br>V. Reactor Containment | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| | defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. | |
| 56 | ~~*Primary*~~ *Containment isolation.*<br>Each line that connects directly to the containment atmosphere and penetrates the ~~primary~~ reactor containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:<br>(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or<br>(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or<br>(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or<br>(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.<br><br>Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word “structure” is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct. | No rationale is provided for the deletion of “primary” from both the title and text of this criterion. Please provide the associated rationale. |
| 57 | *Closed system isolation valves.*<br>Each line that penetrates the ~~primary~~ reactor containment structure and is neither part of the reactor coolant ~~pressure~~ boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be | No comments. |

| Criterion | ARDC Language/ Rationale for Modification | Team Comments |
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| | outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word “structure” is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct.<br><br>Reactor coolant pressure boundary is relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. | |

| DOE Laboratory Team Comments on NRC ARDC<br>VI. Fuel and Reactivity Control | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 60 | Same as GDC<br><br>*Control of releases of radioactive materials to the environment.*<br>The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment. | No comments. |
| 61 | *Fuel storage and handling and radioactivity control.*<br>The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage ~~coolant inventory~~ cooling under accident conditions.<br><br>Rationale<br><br>The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of non-LWR advanced reactors. However, some advanced designs may use dry fuel storage that incorporates cooling jackets that can be liquid-cooled or air-cooled to remove heat. This modification to this GDC allows for both liquid and air-cooling of the dry fuel storage containers. | No comments. |
| 62 | Same as GDC<br><br>*Prevention of criticality in fuel storage and handling.*<br>Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations. | No comments. |

| DOE Laboratory Team Comments on NRC ARDC<br>VI. Fuel and Reactivity Control | | |
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| Criterion | **ARDC Language/<br>Rationale for Modification** | **Team Comments** |
| 63 | Same as GDC<br><br>*Monitoring fuel and waste storage.*<br>Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. | No comments. |
| 64 | *Monitoring radioactivity releases.*<br>Means shall be provided for monitoring the reactor containment atmosphere, ~~spaces containing components for recirculation of loss-of-coolant accident fluids~~, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.<br><br>Rationale<br><br>The phrase "spaces containing components for recirculation of loss of coolant accident fluids" was removed to allow for plant designs that do not have loss-of-coolant accident fluids, but may have other similar equipment that exist in spaces where radioactivity should be monitored. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>I. Overall Requirements | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 1 | Same as GDC<br><br>*Quality standards and records.*<br>Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit. | No comments. |
| 2 | Same as GDC<br><br>*Design bases for protection against natural phenomena.*<br>Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed. | No comments. |
| 3 | Same as ARDC<br><br>*Fire protection.*<br>Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations ~~such as the~~ | DOE's response to NRC's proposed SFR-DC 3 is heavily based on DOE's response to NRC's proposed ARDC 3.<br><br>The term "safety related equipment" in the second sentence of the SFR-DC appears to be redundant to the term "structures, systems, and components important to safety" and potentially changes the applicability of the DC as stated in the first sentence. As written, this text makes a distinction between "safety related equipment" and "structures, systems, and components important to safety" that is made no place else in the design criteria. The words, "safety |

| DOE Laboratory Team Comments on NRC SFR-DC<br>I. Overall Requirements | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | ~~containment and control room~~ with safety- related equipment or structures, systems, and ~~or~~ components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.<br><br>Rationale<br><br>The phrase containing examples where noncombustible and heat resistant materials must be used has been broadened to apply to all advanced reactor designs. | related equipment or" in the second sentence of the proposed SFR-DC should be removed to eliminate the confusion.<br><br>The ITS ambiguity in existing NRC regulations and guidance was identified during the DOE ARDC development. The understood meaning of ITS within the context of 10 CFR 50 Appendix A was defined on page 7 of the DOE report (ML14353A246 and ML14353A248) and was further addressed in DOE's response to NRC Question 40 (ML15204A579). The need for this clarification was also discussed in an NRC public meeting on January 21, 2015 (meeting summary ML15044A081), with DOE indicating that the ARDCs could be heavily impacted if NRC's understanding of the use of the term within Appendix A is different from that provided in the clarifying definition. The NRC's use of ITS in SFR-DC 3 seems to deviate from that understanding.<br><br>Consistency with the GDC is needed with regard to the use of "important to safety" in the ARDC. "Important to safety" is broadly used throughout the GDC; "safety related" is not used anywhere in the GDC. NRC should either confirm that "important to safety" means "safety related" within the context of Appendix A, or explain the difference between the two terms and provide a regulatory basis.<br><br>It is also recommended that the NRC's rational should be reworded as follows: "The phrase containing examples where noncombustible and heat resistant materials must be used has been broadened to apply to SSCs important to safety throughout the unit." |
| 4 | Same as ARDC<br><br>*Environmental and dynamic effects design bases.*<br>Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents~~, including loss of coolant accidents~~. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>I. Overall Requirements | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | the design basis for the piping.<br><br>Rationale<br><br>This change removes the LWR emphasis on loss of cooling accidents (LOCAs) that may not apply to some designs. For example, helium is not needed in a mHTGR to remove heat from the core during postulated accidents and does not have the same importance as water does to LWR designs to assure that fuel integrity is maintained. Therefore, a specific reference to "loss of coolant accidents" is not applicable to all designs. LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design. Reference to pipe whip may not be applicable to designs that operate at low pressure. | |
| 5 | Same as GDC<br><br>*Sharing of structures, systems, and components.*<br>Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>II. Multiple Barriers | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 10 | Same as GDC<br><br>*Reactor design.*<br>The reactor core and associated coolant, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences. | No comments. |
| 11 | Same as ARDC<br><br>*Reactor inherent protection.*<br>The reactor core and associated ~~coolant~~ systems that contribute to reactivity feedback shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.<br><br>Rationale<br><br>The wording has been changed to broaden the applicability from "coolant systems" to additional factors (including structures or other fluids) that may contribute to reactivity feedback. These systems are to be designed to compensate for rapid reactivity increase. | No comments. |
| 12 | Same as ARDC<br><br>*Suppression of reactor power oscillations.*<br>The reactor core and associated structures, coolant, control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.<br><br>Rationale<br><br>The word "structures" was added because items such as reflectors, which could be considered either outside or not part of the reactor core, may affect susceptibility of the core to power oscillations. | No comments. |
| 13 | *Instrumentation and control.*<br>Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions ~~as appropriate~~ to assure | No comments. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | adequate safety, including those variables and systems that can affect the fission process, the integrity of the reactor core**,** the ~~reactor~~ primary coolant ~~pressure~~ boundary, and the containment and its associated systems. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.<br><br>Rationale<br><br>"As appropriate" was removed to provide specificity to the criterion. "Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry. The use of the term "primary" indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system. | |
| 14 | *Primary coolant ~~pressure~~ boundary.*<br>The ~~reactor~~ primary coolant ~~pressure~~ boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry.<br><br>The use of the term "primary" indicates that the SFR-DC is applicable only to the primary cooling system, not the intermediate cooling system.<br><br>The cover gas boundary is included as part of the primary coolant boundary (referred to as RCPB by PRISM) per NUREG-1368 (page 3-38). | No comments. |
| 15 | *~~Reactor~~ Primary coolant system design.*<br>The ~~reactor~~ primary coolant system and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the ~~reactor~~ primary coolant ~~pressure~~ boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.<br><br>Rationale | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>II. Multiple Barriers | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | "Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry.<br><br>The use of the term "primary" indicated that the SFR-DC is applicable only to the primary cooling system, not the intermediate cooling system.<br><br>The cover gas boundary is included as part of the primary coolant boundary (referred to as RCPB by PRISM) per NUREG-1368 (page 3-38). | |
| 16 | *Containment design.*<br>A reactor containment consisting of a high strength, low leakage, pressure retaining structure surrounding the reactor and ~~associated~~ its cooling systems, shall be provided to ~~establish an essentially leak-tight barrier against the uncontrolled~~ control the release of radioactivity to the environment and to assure that the reactor containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.<br><br>The containment leakage shall be restricted to be less than that needed to meet the acceptable onsite and offsite dose consequence limits as specified in 10 CFR Part 50.34 for postulated accidents.<br><br>Rationale<br><br>The Commission approved the staff's recommendation to restrict the leakage of the containment to be less than that needed to meet the acceptable onsite and offsite dose consequence limits [Ref. SRM, SECY-93-092]. Therefore, the Commission agreed that the containment leakage for advanced reactors, similar to and including PRISM, should not be required to meet the "essentially leaktight" statement in GDC16. [Ref: NUREG-1368].<br><br>Also, ARDCs and SFR-DCs 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57 in the DOE report refer to containment in the traditional sense in that these SFR-DCs specify traditional containment systems design, inspection, and testing (including leakage rate testing).<br><br>Furthermore, all past, current, and planned SFR designs use a high strength, low leakage, pressure retaining containment concept which aims to provide a barrier to contain the fission products and other substances and to control the release of radioactivity to the environment. | SFR-DC comment: Change first sentence wording "its cooling systems" to "its primary cooling system." SFR containment designs surround only the primary cooling system. There is no need to include the intermediate loop within the containment since this system will not contain radioactive materials. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>II. Multiple Barriers | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 17 | *Electric power systems.*<br>An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the ~~reactor~~ primary coolant ~~pressure~~ boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.<br><br>The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.<br><br>Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the ~~reactor~~ primary coolant ~~pressure~~ boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a postulated ~~loss-of-coolant~~ accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.<br><br>Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.<br><br>Rationale | DOE's response to NRC's proposed SFR-DC 17 is heavily based on DOE's response to NRC's proposed ARDC 17.<br><br>NRC's proposed SFR-DC 17 retains electric power requirements that may have no safety application in certain SFR reactor designs. As such, this criterion does not apply to many SFR systems.<br><br>Criterion Paragraph 1 requires that, "(2) the core is cooled and containment integrity and other vital functions are maintained…" By specifically encompassing core cooling and containment integrity as well as other vital functions that rely on electric power to the extent needed by an advanced non-LWR reactor design, the text becomes too broad for the SFR-DC. Because SFR-DCs allow for flexible designs, the DOE proposed SFR-DC 17 text should be used to address the requirement.<br><br>Components listed in the criterion (such as batteries and onsite electric distribution systems) are very specific and could impede innovations that mitigate the need for such safety systems if retained as a design requirement. This adherence to LWR electric power system requirements appears to be in conflict the Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" (ML082750370) which summarizes its expectation that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions.<br><br>Advanced reactor design criterion requirements should focus on ensuring that sufficient time and capability is provided following a postulated accident to assure core cooling, containment integrity, and other vital safety functions are maintained. SFR-DC 17 (as currently proposed by NRC) neither recognizes nor credits the unique safety features that will be associated with emerging advanced reactor designs.<br><br>The reference to the ACRS September 24 subcommittee meeting in the rationale should be removed since the subcommittee was addressing the NuScale DSRS and requested the staff to remain flexible in its wording so as to accommodate the possibility that NuScale may not need AC power to accomplish system safety functions after review of their design.<br><br>NRC should adopt the DOE-proposed SFR-DC 17 language. This language focuses on requiring electric power as necessary to ensure safety functions are performed when required. It also provides relief from off-site AC power source |

| DOE Laboratory Team Comments on NRC SFR-DC<br>II. Multiple Barriers | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | The requirements for offsite power are being retained for defense-in-depth considerations. This position was reinforced by a letter from the NRC to Dale Atkinson, Chief Operating Officer, NuScale Power, September 15, 2015 (ML15222A323). At the September 24, 2015 meeting of the Advisory Committee for Reactor Safeguards subcommittee on advanced reactor designs, this subject came up again and the subcommittee was supportive of keeping offsite power requirements in GDC 17 for the NuScale design.<br><br>"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry. The use of the term "primary" indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system. | and on-site safety-grade AC power source requirements for those designs that can successfully accomplish safety functions, assuming a single failure, without needing AC electric power. This approach is flexible, focused, and aligned with guidance contained in the Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" (ML082750370).<br><br>Please note that modifications to SFR-DC 17 to address the needs of advanced non-LWRs also necessitate changes to text contained in SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 18 | Same as GDC<br><br>*Inspection and testing of electric power systems.*<br>Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system.<br><br>Rationale<br><br>GDC 18 is a design- independent companion criterion to GDC 17. | The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency.. |
| 19 | Same as ARDC<br><br>*Control room.*<br>A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions~~including loss-of-coolant accidents~~. Adequate radiation protection shall be provided to | No comments. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent (TEDE) ~~whole body, or its equivalent to any part of the body~~, (TEDE) as defined in § 50.2 for the duration of the accident.<br><br>Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions.<br><br>Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.<br><br>~~Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv (5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.~~<br><br>Rationale<br><br>The criterion was updated to remove specific emphasis on LOCA, which may be not appropriate for advanced designs such as the mHTGR.<br><br>Reference to “whole body, or its equivalent to any part of the body” has been updated to the current TEDE standard as defined in § 50.2. | |

| DOE Laboratory Team Comments on NRC SFR-DC<br>II. Multiple Barriers | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | Control room habitability requirement beyond that associated with radiation protection has been added to address concern that non-radionuclide accidents may also affect control room access and occupancy.<br><br>The last paragraph of the GDC has been eliminated for the ARDC because it is not applicable to future applicants. | |

| DOE Laboratory Team Comments on NRC SFR-DC<br>III. Reactivity Control | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 20 | Same as GDC<br><br>*Protection system functions.*<br>The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety. | No comments. |
| 21 | Same as GDC<br><br>*Protection system reliability and testability.*<br>The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. | No comments. |
| 22 | Same as GDC<br><br>*Protection system independence.*<br>The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function. | No comments. |
| 23 | *Protection system failure modes.*<br>The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, sodium and sodium reaction products, pressure, steam, water, and radiation) are experienced. | In the list of examples, “pressure, steam, water” are shown as added text for SFR-DC 23. However, this text already exists in the corresponding GDC and ARDC. Recommend removing these examples because they do not apply to SFR design protection system failure modes. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>III. Reactivity Control | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | Rationale<br><br>In NUREG-1368, Table 3.3 (page 3-21), (ML063410561) NRC staff recommended adding the phrase "sodium and sodium reaction products" to the list of postulated adverse environments in the GDC. Therefore, "sodium and sodium reaction products" are added to the second list of examples in parenthesis in SFR-DC 23. | |
| 24 | Same as GDC<br><br>*Separation of protection and control systems.*<br>The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired. | No comments. |
| 25 | Same as ARDC<br><br>*Protection system requirements for reactivity control malfunctions.*<br>The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded during any anticipated operational occurrence resulting from a ~~for any~~ single malfunction of the reactivity control systems. ~~, such as accidental withdrawal (not ejection or dropout) of control rods~~<br><br>Rationale<br><br>Text has been added to clarify that the protection system is designed to protect the SAFDLs for AOOs in combination with a single failure; the protection system does not have to protect the SAFDLs during a postulated accident in combination with a single failure. The example was deleted to make ARDC technology neutral. | No comments. |
| 26 | Same as ARDC<br><br>*Reactivity control system redundancy and capability.*<br>At least two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, | 1) The added words "at least" at the beginning should be removed from SFR-DC 26. These words can be perceived as too prescriptive because it seems to imply an additional level of redundancy is required for SFRs.<br><br>2) A comment on ARDC 26 recommends removing the rationale statement |

| DOE Laboratory Team Comments on NRC SFR-DC<br>III. Reactivity Control | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system shall be capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes ~~(including xenon burnout)~~ to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.<br><br>Rationale<br><br>"At least" was added to set a minimum number of independent reactivity control systems; it does not preclude more than two systems.<br><br>The parenthetical phrase "including xenon burnout" has been deleted as it is already addressed by the statement "…rate of reactivity changes resulting from planned, normal power changes." In other words, the second reactivity control system must control the reactivity changes relevant to the specific design for normal plant power changes. This deletion makes the ARDC more technology neutral. For example, xenon burnout does not apply to fast reactor designs.<br><br>"Cold conditions" remains but will have to be defined by a principal design criteria for the specific design. | "Cold conditions" remains but will have to be defined by a principal design criteria for the specific design." If this statement is removed from ARDC 26, it should be retained for SFR-DC 26, because cold conditions are an important consideration for sodium-cooled reactors due to the relatively high melting point of sodium. |
| 27 | Same as ARDC<br><br>*Combined reactivity control systems capability.*<br>The reactivity control systems shall be designed to have a combined capability~~, in conjunction with poison addition by the emergency core cooling system,~~ of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.<br><br>Rationale<br><br>None of the advanced non- LWR designs evaluated in the review utilized poison addition via an ECCS. | DOE agrees with the NRC rationale for SFR-DC 27. However, the NRC staff rationale presented for SFR-DC 27 is inconsistent with the NRC staff position to retain SFR-DC 35. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>III. Reactivity Control | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | In addition, ARDC 34, *Residual heat removal*, combines the ECCS requirements in GDC 35 into ARDC 34, because none of the advanced non-LWR designs evaluated utilized an ECCS. Advanced non-LWR designs that do use poison addition or an ECCS will have to look to GDC 27 and GDC 35 for guidance. | |
| 28 | *Reactivity limits.*<br>The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the primary ~~reactor~~ coolant boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor ~~pressure~~ vessel internals to impair significantly the capability to cool the core. ~~These postulated reactivity accidents shall include consideration of [rod ejection (unless prevented by positive means), rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition].~~<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry. The use of the term "primary" indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system.<br><br>The list of "postulated reactivity accidents" has been deleted. Each design will have to determine its postulated reactivity accidents based on the specific design and associated risk evaluation. | No comments. |
| 29 | Same as GDC<br><br>*Protection against anticipated operational occurrences.*<br>The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>IV. Fluid Systems | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 30 | *Quality of ~~reactor~~ primary coolant ~~pressure~~ boundary.* Components which are part of the ~~reactor~~ primary coolant ~~pressure~~ boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical.<br>Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry.<br><br>The use of the term "primary" indicates that the SFR-DC is applicable only to the primary cooling system, not the intermediate cooling system.<br><br>The cover gas boundary is included as part of the reactor primary coolant boundary (referred to as RCPB by PRISM) per NUREG-1368 (page 3-38). | No comments. |
| 31 | *Fracture prevention of ~~reactor~~ primary coolant ~~pressure~~ boundary.*<br>The ~~reactor~~ primary coolant ~~pressure~~ boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry.<br><br>The use of the term "primary" indicates that the SFR-DC is applicable only to the primary cooling system, not the intermediate cooling system.<br><br>The cover gas boundary is included as part of the reactor primary coolant boundary (referred to as RCPB by PRISM) per NUREG-1368 (page 3-38). | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>IV. Fluid Systems | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 32 | *Inspection of ~~reactor~~ primary coolant ~~pressure~~ boundary.*<br>Components which are part of the ~~reactor~~ primary coolant ~~pressure~~ boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor ~~pressure~~ vessel.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to conform to standard terms used in the LMR industry.<br><br>The use of the term "primary" indicates that the SFR-DC is applicable only to the primary cooling system, not the intermediate cooling system.<br><br>The cover gas boundary is included as part of the reactor primary coolant boundary (referred to as RCPB by PRISM) per NUREG-1368 (page 3-38).<br><br>The staff modified the LWR GDC by replacing the term "reactor pressure vessel" with "reactor vessel", which staff believes is a more generically applicable term. | No comments. |
| 33 | *~~Reactor~~ Primary coolant inventory maintenance~~makeup~~.*<br>A system to maintain ~~supply~~ ~~reactor~~ primary coolant inventory ~~makeup~~ for protection against small breaks in the ~~reactor~~ primary coolant ~~pressure~~ boundary shall be provided~~. The system safety function shall be~~ as necessary to assure that specified acceptable fuel design limits are not exceeded as a result of ~~reactor~~ primary coolant inventory loss due to leakage from the ~~reactor~~ primary coolant ~~pressure~~ boundary and rupture of small piping or other small components which are part of the boundary. The system shall be designed to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps, and valves used to maintain primary coolant inventory during normal reactor operation.<br><br>Rationale | DOE's response to NRC's proposed SFR-DC 33 is heavily based on DOE's response to NRC's proposed ARDC 33.<br><br>Recognizing that the inventory control system may be unnecessary for some designs to maintain safety functions that assure fuel design limits are not exceeded, criterion sentences 1 and 2 should delete the words, "The system safety function shall be…", and be combined to read: "A system to maintain primary coolant inventory for protection against small breaks in the primary coolant boundary shall be provided as necessary to assure that the …".<br><br>The comment brings NRC SFR-DC 33 text back into alignment with DOE-proposed wording. For low pressure advanced reactor primary systems, small breaks are not credible sources of significant coolant inventory loss in a short time period.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>IV. Fluid Systems | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | "Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to reflect that the SFR primary system operates at low-pressure and to conform to standard terms used in the LMR industry.<br><br>The coolant boundary design requirements differ from the traditional LWR coolant pressure boundary requirements. The effects of low pressure design are acknowledged in NUREG-1368 (page 3-28) (ML063410561) under discussion of GDC 4 and on (page 3-30) under GDC 14. The use of the term "primary" implies the GDC is applicable to the primary cooling system, not the intermediate cooling system.<br><br>Both pool- and loop-type SFR designs limit loss of primary coolant so that an inventory adequate to perform the safety function of the residual heat removal system is maintained under operating, maintenance, testing, and postulated accident conditions. | |
| 34 | *Residual heat removal.*<br>A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the ~~The~~ system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable fuel design limits and the design conditions of the ~~reactor~~ primary coolant boundary are not exceeded.<br><br>During postulated accidents, the system safety function shall transfer heat from the reactor core at a rate such that fuel and clad damage that could interfere with continued effective cooling is prevented, sodium boiling is precluded, and the design conditions of the primary coolant boundary are not exceeded.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>A passive boundary shall separate primary coolant from the working fluid of the residual heat removal system and any fluid in the residual heat removal system that is separated from the primary coolant by a single passive barrier shall not be chemically reactive with the primary coolant. In addition, the working fluid of residual heat removal system shall be at a | The phrase "sodium boiling is precluded" should be removed. During postulated accidents, localized boiling may occur, and is acceptable, under the definition of continued effective core cooling.<br><br>In the fourth paragraph, the phrase "…shall not be chemically reactive with the primary coolant…" should be replaced with the phrase "…shall be chemically compatible with the primary coolant…" Requiring that the RHRS coolant shall be chemically nonreactive with sodium can be open to misinterpretation. Instead, requiring that RHRS coolant to be chemically compatible with sodium is more appropriate. For example, NaK (a possible RHRS coolant) is chemically compatible with sodium; however, it is not chemically nonreactive with sodium.<br><br>The requirement for the RHRS to have sufficient capacity to maintain design conditions of the primary coolant boundary as stated in NRC's proposed paragraph is valid for SFRs.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | higher pressure than the primary coolant system.<br><br>Rationale<br><br>SFR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35.<br><br>“Ultimate heat sink” has been added to clarify that if SFR-DC 44 is deemed not applicable to the design, the RHR system is then required to provide the heat removal path to the ultimate heat sink.<br><br>“Reactor coolant pressure boundary” has been relabeled as “primary coolant boundary” to reflect that the SFR primary system operates at low-pressure and to conform to standard terms used in the LMR industry. The use of the term “primary” indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system.<br><br>The second paragraph was added to clarify that the safety function of the residual heat removal system during postulated accidents is to provide continuous effective core cooling. For SFRs, that cooling is provided at a rate sufficient to prevent propagation of fuel failures. The last phrase was added to the paragraph to assure that residual heat removal capability is sufficient to maintain the integrity of the primary coolant boundary during postulated accidents.<br><br>A paragraph from NUREG- 1368 (page 3-41) was added describing the characteristics of the residual heat removal working fluid and its associated operating pressure. A single passive barrier is adequate defense in depth when the residual heat removal working fluid is not chemically reactive with the primary coolant. If chemically reactive at least two passive barriers must separate the two systems. The higher pressure requirement is to ensure any leakage in the interface between the two systems does not result in a release of radioactive primary coolant to the non-radioactive part of the heat transport system. | |
| 35 | Same as ARDC<br><br>*Emergency core cooling.*<br>If the system as described in ARDC 34 does not provide continuous effective core cooling during postulated accidents and does not assure that the design conditions of the reactor coolant boundary are preserved; | DOE’s response to NRC’s proposed SFR-DC 35 is heavily based on DOE’s response to NRC’s proposed ARDC 35.<br><br>DOE did not propose ARDC or SFR-DC content, but instead proposed in the ARDC that; "If a separate ECCS system is required for an advanced reactor, the PDC process for that reactor must look directly to GDC 35 for guidance." |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | then a~~A~~ system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant such that continuous effective core cooling is maintained.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>In most advanced reactor designs, residual heat removal is addressed by ARDC 34. If the design is such that ARDC 34 is not adequate to ensure residual heat removal under normal operations and postulated accidents then additional system(s) are required and would be addressed by this ARDC 35 to ensure continuous effective core cooling. | This proposal also applies to the SFRs. This does not suggest an advanced reactor design equivalent to GDC 35 is needed given that the proposed SFR-DC 35 does not support any known SFR technology. If a system described in SFR-DC 34 does not provide continuous effective core cooling during postulated accidents (which is required), the system is improperly designed and subject to revision to meet the SFR-DC 34 requirement. A "contingency" system should not be assumed as an arbitrary design option. Criterion 35 is obsolete and not applicable to SFRs.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 36 | Same as ARDC<br><br>*Inspection of ~~emergency core~~ ~~cooling~~ residual heat removal system.*<br>The ~~emergency core cooling~~ ~~system~~ residual heat removal shall be designed to permit appropriate periodic inspection of important components~~, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping,~~ to assure the integrity and capability of the system.<br><br>Rationale<br><br>Title has been renamed and GDC revised to provide for inspection of the residual heat removal systems as required for ARDC 34.<br><br>The example list has been deleted because it applies to LWR designs and each specific design will have different important components associated with residual heat removal. This revision allows for a technology neutral ARDC.<br><br>Review of the proposed DOE SFR and HTGR DCs found that only SFR provided specific examples of important components but were generic in | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>IV. Fluid Systems | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | nature and did not add any significant additional guidance | |
| 37 | Same as ARDC<br><br>*Testing of residual heat removal ~~emergency core~~ ~~cooling~~ system*.<br>The residual heat removal ~~emergency core cooling~~ system shall be designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of associated systems and interfaces with an ultimate heat sink including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.<br><br>Rationale<br><br>GDC 37 system has been renamed and revised to provide for testing of the residual heat removal system of ARDC 34.<br><br>A specific requirement for pressure and leaktight testing was retained in the ARDC as future advance designs may employ pressure retaining RHR designs. If the applicable system in the advanced design is not pressure retaining, then "periodic pressure testing" and "leaktight integrity" could be removed in the specific design criteria.<br><br>"Active" has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature.<br><br>Reference to operation of applicable portions of the protection system, cooling water system, and power transfers is considered part of the more general "associated systems." Together with the ultimate heat sink, they are part of the operability testing of the system as a whole. | DOE's response to NRC's proposed SFR-DC 37 is heavily based on DOE's response to NRC's proposed ARDC 37.<br><br>SFRs do not operate at pressure. The first sentence of the criterion should therefore be changed from "…periodic pressure and functional testing…" to "…periodic functional testing…" Functional testing can include pressure testing as appropriate and the required pressure capability specified in test acceptance criteria.<br><br>Similarly, the phrase under (1) should be changed from "the structural and leaktight integrity of its components," to "the structural integrity of its components." Structural integrity can include leaktightness as appropriate and the degree of leaktightness specified in test acceptance criteria.<br><br>The last sentence of the criterion states "…including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system…" which should be deleted. The NRC criterion rationale notes that operation of applicable portions of the protection system, cooling water system, and power transfers is considered part of the more general "associated systems" and together with the ultimate heat sink, they are part of the operability testing of the system as a whole. This suggests the criterion statement was intended for deletion.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 38 | Same as ARDC<br><br>*Containment heat removal.*<br>A system to remove heat from the reactor containment shall be provided as necessary ~~The system safety function shall be~~ to maintain ~~reduce~~ | The basis for adding "…including electric power systems" to Criteria 38 and 41 is unclear and not addressed in the rationale. Electric power systems are included under "components and features," thereby making the new text unnecessary. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | ~~rapidly, consistent with the functioning of other associated systems,~~ the containment pressure and temperature within acceptable limits following ~~following any loss-of-coolant~~ postulated accidents. ~~and maintain them at acceptably low levels.~~<br><br>Suitable redundancy in components and features, including electric power systems , and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>"…as necessary…" is meant to condition ARDC 38 application to designs requiring heat removal for conventional containments which are found to require heat removal measures.<br><br>LOCA reference has been removed to provide for any postulated accident that might affect the containment structure.<br><br>Containment structure safety system redundancy is addressed in second paragraph. | The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 39 | Same as ARDC<br><br>*Inspection of containment heat removal system*.<br>The containment heat removal system shall be designed to permit appropriate periodic inspection of important components~~, such as the torus, sumps, spray nozzles, and piping~~ to assure the integrity and capability of the system.<br><br>Rationale<br><br>Examples were deleted to make the ARDC technology neutral. | No comments. |
| 40 | Same as ARDC<br><br>*Testing of containment heat removal system.*<br>The containment heat removal system shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the | The words under Criterion Item (3), "…including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system," are addressed "…under operation of associated systems" and should be deleted. The last paragraph of the NRC rationale suggests the statement was intended |

| DOE Laboratory Team Comments on NRC SFR-DC<br>IV. Fluid Systems | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | structural ~~and leak-tight~~ integrity of its components, (2) the operability and performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the system as a whole, and under conditions as close to the design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system, including operation of associated systems.<br><br>Rationale<br><br>Specific mention of "pressure" testing has been removed yet remains a potential requirement should it be necessary as a component of "…appropriate periodic functional testing..." of cooling systems.<br><br>"Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability.<br>Reference to operation of applicable portions of the protection system, cooling water systems, and power transfers is considered part of the more general "associated systems" for operability testing of the system as a whole. | for deletion. This revision makes the criterion more technology neutral.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 41 | Same as ARDC<br><br>*Containment atmosphere cleanup.*<br>Systems to control fission products ~~hydrogen, oxygen~~ and other substances which may be released into the reactor containment shall be provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of ~~hydrogen or oxygen and~~ other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.<br><br>Each system shall have suitable redundancy in components and features, including electric power systems, and suitable interconnections, leak detection, isolation, and containment capabilities to assure ~~that~~ that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure. | The basis for adding "…including electric power systems" to Criteria 38 and 41 is unclear and not addressed in the rationale. Electric power systems are included under "components and features," thereby making the new text unnecessary.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | Rationale<br><br>Advanced reactors offer potential for reaction product generation that is different from that associated with clad metal-water interactions. Therefore, the terms “hydrogen” and “oxygen” are removed while “other substances” is retained to allow for exceptions. | |
| 42 | Same as GDC<br><br>*Inspection of containment atmosphere cleanup systems.*<br>The containment atmosphere cleanup systems shall be designed to permit appropriate periodic inspection of important components, such as filter frames, ducts, and piping to assure the integrity and capability of the systems. | No comments. |
| 43 | Same as ARDC<br><br>*Testing of containment atmosphere cleanup systems.*<br>The containment atmosphere cleanup systems shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the structural ~~and leak-tight~~ integrity of its components, (2) the operability and performance of the ~~active~~ system components, ~~of the systems such as fans, filters, dampers, pumps, and valves~~ and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and including the operation of associated systems.<br><br>Rationale<br><br>“Active” has been deleted in item (2) as appropriate operability and performance testing of system components is required regardless of active or passive nature, as are cited examples of active system components.<br><br>Examples of active systems under item (2) have been deleted both to conform to similar wording in ARDC 37 and 40 and ensure passive as well as active system components are considered. | Criterion Item (3): The words “operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and” are included under “including operation of associated systems” and should be deleted. This makes the criterion more technology neutral.<br><br>The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | Specific mention of "pressure" testing has been removed yet remains a potential requirement should it be necessary as a component of "…appropriate periodic functional testing…" of cooling systems.<br><br>"Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability. | |
| 44 | Same as ARDC<br><br>*Structural and equipment cooling ~~Cooling water~~.*<br>In addition to the heat rejection capability of the residual heat removal system, ~~A~~ systems to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided, as necessary. ~~The system safety function shall be~~ to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor design system differences to include safety-related cooling requirements for SSCs, if applicable; this ARDC does not address the residual heat removal system required under ARDC 34. | The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |
| 45 | Same as ARDC<br><br>*Inspection of structural and equipment cooling ~~water~~ systems.*<br>The ~~cooling water~~ structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the systems.<br><br>Rationale | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>IV. Fluid Systems | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs. | |
| 46 | Same as ARDC<br><br>*Testing of structural and equipment cooling ~~. water~~ systems.*<br>The structural and equipment cooling ~~water~~ systems shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the structural ~~and leaktight~~ integrity of their ~~its~~ components, (2) the operability and the performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring~~s~~ the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems. ~~and for loss-of-coolant accidents, including operation of~~ and applicable portions of the protection system and the transfer between normal and emergency power sources.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs.<br><br>Specific mention of “pressure” testing has been removed yet remains a potential requirement should it be necessary as a component of “…appropriate periodic functional testing...” of cooling systems.<br><br>“Leaktight” integrity would be demonstrated through appropriate functional testing of system performance and operability.<br><br>“Active” has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature.<br><br>LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs. | The comment with SFR-DC 17 notes that change to SFR-DC 17 will require review and adaptation to SFR-DCs 18, 33, 34, 35, 37, 38, 40, 41, 43, 44, and 46 to ensure consistency. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>V. Reactor Containment | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 50 | Same as ARDC<br><br>*Containment design basis.*<br>The reactor containment structure, including access openings, penetrations, and the containment heat removal system shall be designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and with sufficient margin, the calculated pressure and temperature conditions resulting from postulated accidents. ~~any loss-of-coolant accident~~. This margin shall reflect consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as fission products, potential spray or aerosol formation, and potential exothermic chemical reactions ~~energy in steam generators and as required by § 50.44 energy from metal-water and other chemical reactions that may result from degradation but not total failure of emergency core cooling functioning~~, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.<br><br>Rationale<br><br>ARDC-50 specifically addresses a containment structure in the opening sentence and ARDCs 51-57 support the containment structure's design basis. Therefore, ARDC 51 – 57 are modified by adding the word "structure" to highlight the containment structure-specific criteria.<br><br>The phrase "loss of coolant accident" is LWR-specific because this is understood to be the limiting containment structure accident for an LWR design. It is replaced by the phrase "postulated accident" to allow for consideration of the design-specific containment structure limiting accident for advanced non- LWR designs.<br><br>The example at the end of subpart 1 of the ARDC is LWR-specific and therefore deleted. | There is no objection to the wording of SFR-DC 50. The corresponding rationale for ARDC 50 notes the deletion of the LWR-specific example in subpart 1 of the ARDC. However, the rationale does not address the selection of the replacement examples selected for inclusion in ARDC 50. Subsequently, the SFR-DC 50 is noted to be the same as the ARDC. The source for the revised example list is actually found in NUREG-1368, "Preapplication Safety Evaluation report for PRISM LMR," which is specific to SFR designs. On page 3-50 of the NUREG, the NRC staff recommended replacing the existing example with "fission products, potential spray or aerosol formation, and potential exothermic chemical reactions." This source material for the text change is not currently identified in the ARDC rationale and there is no SFR-DC rationale. For clarity and consistency, the ARDC 50 rationale should reflect the source of the text change and why it is generically reflective of all non-LWR designs or leave design-specific text examples to the design-specific DCs with supporting rationale. |
| 51 | Same as ARDC<br><br>*Fracture prevention of containment pressure boundary.*<br>The ~~reactor containment~~ boundary of the reactor containment structure shall be designed with sufficient margin to assure that under operating, | Recommend deleting the word "pressure" in the SFR-DC criterion title. An SFR containment is a boundary/barrier to the release of radioactivity and not a pressure boundary. Deleting the word "pressure" in the SFR-DC title provides clarity for SFR designs. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | maintenance, testing, and postulated accident conditions (1) its ~~ferritic~~ materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining<br>(1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC- 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word "structure" is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct.<br><br>The term "ferritic" was removed in order to not limit the scope of the criterion to ferritic materials. With this revision, the staff believes that this criterion is generically applicable to all non-LWR designs. | |
| 52 | Same as ARDC<br><br>*Capability for containment leakage rate testing*.<br>The reactor containment structure and other equipment which may be subjected to containment test conditions shall be designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word "structure" is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>V. Reactor Containment | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 53 | Same as ARDC<br><br>*Provisions for containment testing and inspection.*<br>The reactor containment structure shall be designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations,(2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak-tightness of penetrations which have resilient seals and expansion bellows.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word "structure" is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct. | No comments. |
| 54 | *Piping systems penetrating containment.*<br>Piping systems penetrating the ~~primary~~ reactor containment structure shall be provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities necessary to perform the containment safety function and which reflect the importance to safety of preventing radioactivity releases from containment through ~~isolating~~ these piping systems. ~~Such piping~~ Piping systems shall be designed with ~~a~~ the capability to verify by testing ~~periodically~~ the ~~operability of the~~ operational readiness of any isolation valves and associated apparatus periodically, ~~and to~~ ~~determine if~~ and to confirm that valve leakage is within acceptable limits.<br><br>Rationale<br><br>The word "structure" was added to this SFR-DC to clearly convey the understanding that this criterion only applies to designs employing containment structures. In some cases, the make the phrase grammatically correct.<br><br>Not all penetrations will provide a release path to the atmosphere. Piping that may be of interest in the case of an SFR design is for the intermediate heat transport system (IHTS) and the passive residual heat removal system. Based on stakeholder input, a designer may be able to | 1) The last sentence of the criterion should not incorporate the AMSE Standard language. Although it is recognized that the proposed revisions are intended to align the language with the associated ASME Standard, that Standard is already incorporated by reference, thus confirming the technical expectation. Changing the criterion text in this pending Regulatory Guide could create future inconsistencies between the criterion and the ASME Standard, if the Standard is updated and endorsed by the NRC in the future.<br><br>2) The last sentence of the criterion should include a lead-in phrase, such as "When isolation valves are required…" to complement the revisions in the first sentence of the criterion that provide for added flexibility in meeting this design criterion. The first sentence provides the flexibility to establish a safety case that demonstrates that containment isolation valves are not required, but the second sentence appears to return to a more rigid requirement for containment isolation valves. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | satisfactorily demonstrate that containment isolation valves are not required for an SFR design. This rewording for the SFR-DC provides a designer the opportunity to present the safety case without containment isolation valves and associated need for testing. Otherwise, NUREG-1368 (ML063410561) (page 3-51) indicated that GDC 54 was applicable as written.<br><br>ANSI/ANS-54.1-1989 recommended revising the phrase "…containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems." to "…containment capabilities as required to perform the containment safety function."<br><br>The adjustment to the last sentence enhances the clarity of the sentence with respect to the latest terminology used for valve periodic verification and operational readiness. It also removes the introductory statement, as the definition of "required" could be confusing— the designer will present the safety case for what is necessary, and the NRC staff will review it.<br><br>The ASME Operation and Maintenance of Nuclear Power Plants, Division 1: OM Code: Section IST (ASME OM Code) defines operational readiness as the ability of a component to perform its specified functions. The ASME OM Code is incorporated by reference in the NRC regulations in 10 CFR 50.55a, including the definition of operational readiness for pumps, valves, and dynamic restraints. | |
| 55 | ~~*Reactor*~~ *Primary coolant* ~~*pressure*~~ *boundary penetrating containment.*<br>Each line that is part of the ~~reactor~~ primary coolant ~~pressure~~ boundary and that penetrates the ~~primary~~ reactor containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:<br>(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or<br>(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or<br>(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or<br>(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the | 1) No rationale is provided for the deletion of "as necessary" from the text of this criterion. This deletion does not appear to be needed, and the existing GDC "as necessary" text should be retained. The deletion could be viewed by some as a reduction in current levels of flexibility.<br><br>2) The content of the second, third, and fourth paragraphs of the Rationale may be difficult for readers to associate with the updated criterion without a more expanded discussion of the typical SFR configuration. It is suggested that the Rationale be further expanded to support "cold reader" understanding, consistent with NRC's Question #13 and the DOE response provided on July 15, 2015. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | automatic isolation valve outside containment.<br><br>Isolation valves outside containment shall be located as close to containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.<br><br>Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them shall be provided ~~as necessary~~ to assure adequate safety.<br><br>Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, shall include consideration of the population density, use characteristics, and physical characteristics of the site environs.<br><br>Rationale<br><br>The word "structure" was added to this SFR-DC to clearly convey the understanding that this criterion only applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct.<br><br>The title of SFR-DC 55 is the *"Primary coolant boundary penetrating containment."* The SFR intermediate loop is a separate closed system that does not allow any direct mixing of intermediate fluid with the primary coolant sodium.<br><br>The tubing of the IHX and associated intermediate loop piping inside the RV are a part of the primary coolant boundary. SFR-DC 57, "*Closed system isolation valves,"* addresses closed systems that penetrate containment and would be the appropriate place to address a closed system, such as an intermediate loop, that penetrates containment and is not part of the primary coolant boundary (in its entirety). This is similar to the treatment of the main steam system and the steam generator in a PWR. | |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | "Reactor coolant pressure boundary" has been relabeled as "primary coolant boundary" to reflect that the SFR primary system operates at low-pressure and to conform to standard terms used in the LMR industry. The use of the term "primary" implies the SFR- DC is applicable to the primary cooling system, not the intermediate cooling system. | |
| 56 | Same as ARDC<br><br>~~*Primary*~~ *Containment isolation.*<br>Each line that connects directly to the containment atmosphere and penetrates the ~~primary~~ reactor containment structure shall be provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:<br>(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or<br>(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or<br>(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or<br>(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.<br><br>Isolation valves outside containment shall be located as close to the containment as practical and upon loss of actuating power, automatic isolation valves shall be designed to take the position that provides greater safety.<br><br>Rationale<br><br>ARDCs 51-57 support ARDC 50, which specifically applies to advanced non-LWR designs that utilize a fixed containment structure. Therefore, the word "structure" is added to each of these ARDCs to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word "the" was also added to make the phrase grammatically correct. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>V. Reactor Containment | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 57 | *Closed system isolation valves.*<br>Each line that penetrates the ~~primary~~ reactor containment structure and is neither part of the ~~reactor~~ primary coolant ~~pressure~~ boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve ~~which~~ unless it can be demonstrated that the containment safety function can be met without an isolation valve and assuming failure of a single active component. The isolation valve, if ~~which shall~~ required, shall be either automatic, or locked closed, or capable of remote manual operation. This valve shall be outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.<br><br>Rationale<br><br>The word “structure” was added to this SFR-DC to clearly convey the understanding that this criterion applies to designs employing containment structures. In some cases, the word “the” was also added to make the phrase grammatically correct.<br><br>Reactor coolant pressure boundary” has been relabeled as “primary coolant boundary” to reflect that the SFR primary system operates at low-pressure and to conform to standard terms used in the LMR industry. The use of the term “primary” implies the SFR- DC is applicable to the primary cooling system, not the intermediate cooling system. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>VI. Fuel and Radioactivity Control | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 60 | Same as GDC<br><br>*Control of releases of radioactive materials to the environment.*<br>The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment. | No comments. |
| 61 | Same as ARDC<br><br>*Fuel storage and handling and radioactivity control.*<br>The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage ~~coolant inventory~~ cooling under accident conditions.<br><br>Rationale<br><br>The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of non-LWR advanced reactors. However, some advanced designs may use dry fuel storage that incorporates cooling jackets that can be liquid-cooled or air-cooled to remove heat. This modification to this GDC allows for both liquid and air-cooling of the dry fuel storage containers. | No comments. |
| 62 | Same as GDC<br><br>*Prevention of criticality in fuel storage and handling.*<br>Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>VI. Fuel and Radioactivity Control | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | configurations. | |
| 63 | Same as GDC<br><br>*Monitoring fuel and waste storage.*<br>Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. | No comments. |
| 64 | *Monitoring radioactivity releases.*<br>Means shall be provided for monitoring the reactor containment atmosphere, spaces containing components for ~~recirculation of loss-of-coolant accident fluids~~ primary system sodium and cover gas cleanup and processing, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.<br><br>Rationale<br><br>In NUREG-1368, Table 3.3 (page 3-25) (ML063410561) NRC staff recommended deleting the GDC-64 phrase "spaces containing components for recirculation of loss-of- coolant accident fluids." Otherwise, the NRC staff noted that criterion requirements are independent of the design of SFRs (page 3-55).<br><br>Text was added to identify other SFR plant areas that should also be included to maintain consideration of all areas subject to monitoring. Therefore, primary system sodium and cover gas cleanup systems that may be outside containment and effluent processing systems are considered in place of the current text. | No comments. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>VII. Additional SFR-DC | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 70 | *Intermediate coolant system.*<br>An intermediate cooling system shall be provided. A single passive barrier shall separate intermediate coolant from primary coolant; at least a single passive barrier shall separate the energy conversion system coolant from intermediate coolant. The intermediate coolant shall be chemically nonreactive with sodium. A pressure differential shall be maintained across the primary to intermediate barrier such that any coolant barrier leakage would flow from the intermediate coolant system to the primary coolant system. The intermediate coolant boundary shall be designed to permit the conduct of a surveillance program and inspection in areas where intermediate coolant leakage out of the intermediate coolant system, or energy conversion system coolant leakage into the intermediate coolant system, may hinder or prevent a structure, system, or component from performing any of its intended safety functions.<br><br>Rationale<br><br>NRC considered the DOE's proposed SFR-DC 70 and made changes based on the "Response to NRC Staff Questions on the U.S. Department of Energy Report, "Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors" (ML15204A579) (pages 8-11) NUREG-1368 (page 3-57) (ML063410561) Section 3.2.4.5 suggested the need for a separate criterion for the intermediate coolant system. Also separate criteria were included in NUREG-0968 (ML082381008) (Criterion 31– Design of Intermediate Cooling System and Criterion 33–Inspection of Intermediate Cooling System). | DOE recommends rephrasing sentences two and three as follows:<br><br>"*If separated from the primary coolant by a single passive barrier, the intermediate coolant shall be chemically compatible with sodium. At least a single passive barrier shall separate the energy conversion system coolant from intermediate coolant.*"<br><br>Requiring that "the intermediate coolant shall be chemically nonreactive with sodium" can be open to misinterpretation. Instead, requiring that "intermediate coolant to be chemically compatible with sodium" is more appropriate. For example, NaK (a possible intermediate coolant) is chemically compatible with sodium; however, it is not chemically nonreactive with sodium.<br><br>Also, NRC wording creates an impression that this criterion mandates the use of a single passive barrier between primary and intermediate coolants (thus rules out the use of double-wall IHX tubes). |
| 71 | *Primary coolant & cover gas purity control.*<br>Systems shall be provided as necessary to maintain the purity of primary coolant sodium and cover gas within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radionuclide concentrations.<br><br>Rationale<br><br>NRC considered the DOE's proposed SFR-DC 71 and made changes based on the "Response to NRC Staff Questions on the U.S. Department of Energy Report, "Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors" (ML15204A579) (pages 12-13).<br><br>NUREG-1368 (page 3-57) (ML063410561) Section 3.2.4.6 suggested the | No comments. |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | need for a separate criterion for sodium and cover gas purity control. Also a separate criterion was included in NUREG-0968 (ML082381008) (Criterion 34– Reactor and intermediate coolant and cover gas purity control). | |
| 72 | *Sodium heating systems.*<br>Heating systems shall be provided for systems and components important to safety, which contain or could be required to contain sodium. These heating systems and their controls shall be appropriately designed to assure that the temperature distribution and rate of change of temperature in systems and components containing sodium are maintained within design limits assuming a single failure. If plugging of any cover gas line due to condensation or plate out of sodium aerosol or vapor could prevent accomplishing a safety function, the temperature control associated with that line shall be considered important to safety.<br><br>Rationale<br><br>NRC considered the DOE's proposed SFR-DC 72 and made changes based on the "Response to NRC Staff Questions on the U.S. Department of Energy Report, "Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors" (ML15204A579) (pages 13-14).<br><br>NUREG-1368 (page 3-56) (ML063410561) Section 3.2.4.2 suggested the need for a separate criterion for sodium heating system. Also, a separate criterion was included in NUREG-0968 (ML082381008) (Criterion–7 Sodium Heating Systems). | The last sentence (regarding plugging of a cover gas line) does not belong in SFR-DC 72 that is about sodium heating systems.<br><br>Cover gas purity control is addressed in SFR-DC 71 (fouling and plugging of passages is specifically covered). |
| 73 | *Sodium leakage detection and reaction prevention and mitigation*.<br>Means to detect sodium leakage and to limit and control the extent of sodium-air and sodium-concrete reactions and to extinguish fires resulting from these sodium-air and sodium-concrete reactions shall be provided to assure that the safety functions of structures, systems and components important to safety are maintained. Special features such as inerted enclosures or guard vessels shall be provided for systems containing sodium.<br><br>Rationale<br><br>NRC considered the DOE's proposed SFR-DC 73 and made changes based on the "Response to NRC Staff Questions on the U.S. Department of Energy Report, "Guidance for Developing Principal Design Criteria for | 1) DOE suggests including the words "as necessary" after "…shall be provided" as originally suggested in the DOE version of SFR-DC 73.<br><br>Means to detect sodium leakage and to limit and control the sodium fires are needed "as necessary" (only to assure that the safety functions of SSCs important to safety are maintained). A non-radioactive sodium leakage, unless it interferes with an SSC performing its intended safety function, is considered an operational concern, not a safety issue.<br><br>2) DOE also suggests deleting the phrase "and to extinguish fires resulting from these sodium-air and sodium-concrete reactions."<br><br>Requiring "to extinguish" any sodium fire is too prescriptive since "suppression" can be the preferred method when dealing with some sodium |

| Criterion | SFR-DC Language/ Rationale for Modification | Team Comments |
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| | Advanced Non-Light Water Reactors" (ML15204A579) (pages 15-16).<br><br>NUREG-1368 (page 3-56) (ML063410561) Section 3.2.4.1 suggested the need for a separate criterion for protection against sodium reactions. Also, a separate criterion was included in NUREG-0968 (ML082381008) (Criterion–4 Protection against Sodium and NaK reactions). | leaks. The lead-in part of the first sentence that requires to "limit and control" the sodium fires is acceptable as it encompasses both functions to extinguish or suppress. |
| 74 | *Sodium/water reaction prevention/mitigation.*<br>Structures, systems, and components containing sodium shall be designed and located to limit the adverse effects of chemical reactions between sodium and water on the capability of any structure, system, or component to perform any of its intended safety functions. Means shall be provided to limit contact between sodium and water such that chemical reactions between sodium and water will not affect the capability of any structure, system, or component to perform any of its intended safety functions.<br><br>To prevent loss of any plant safety function, the sodium-steam generator system shall be designed to detect and contain sodium-water reactions and limit the effects of the energy and reaction products released by such reactions, as well as to extinguish a fire as a result of such reactions.<br><br>Rationale<br><br>NRC considered the DOE's proposed SFR-DC 74 and made changes based on the "Response to NRC Staff Questions on the U.S. Department of Energy Report, "Guidance for Developing Principal Design Criteria for Advanced Non-Light Water Reactors" (ML15204A579) (pages 16-18) NUREG-1368 (page 3-56) (ML063410561) Section 3.2.4.1 suggested the need for a separate criterion for protection against sodium reactions. Also, a separate criterion was included in NUREG-0968 (ML082381008) (Criterion–4 Protection against Sodium and NaK reactions). Fire considerations are added for consistency with SFR-DC 73. | 1) The second sentence seems to be largely redundant with the first sentence. The following combined form is advised instead:<br><br>*Structures, systems, and components containing sodium shall be designed and located to avoid contact between sodium and water, and to limit the adverse effects of chemical reactions between sodium and water on the capability of any structure, system, or component to perform any of its intended safety functions.*<br><br>2) The last portion of the last paragraph ("*as well as to extinguish a fire as a result of such reactions"*) is too prescriptive for design of a steam generator, and it should be deleted. Extinguishing a fire would not be a safety function for a steam generator<br><br>Fire protection is already addressed in SFR-DC 3. Also, suppression, not extinguishing, may be the preferred method when dealing with sodium-fires and, when sodium-water reactions occur, hydrogen as its byproduct is often deliberately set fire in a flare-stack (to prevent its accumulation). |
| 75 | *Quality of the intermediate coolant boundary.*<br>Components which are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.<br><br>Rationale | The proposed new criteria 75, 76, and 77 should be removed. Quality of the intermediate coolant boundary within the context of a leakage that may prevent a structure, system, or component from performing any of its intended safety functions is already addressed in SFR-DC 70. More general considerations for sodium leakage detection, reaction prevention and mitigation are also captured in SFR-DC 73. If, for some designs, the intermediate coolant system is also utilized as a path for decay heat removal, then the quality of its boundary is covered in SFR-DC 34, 36, and 37. |

| DOE Laboratory Team Comments on NRC SFR-DC<br>VII. Additional SFR-DC | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | This criterion is unique to the SFR design because, based on the information available to the staff, it is the only nuclear plant design for which there is an intermediate coolant loop. This criterion is identical to GDC 30 in 10 CFR 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed , fabricated, and tested using quality standards and controls sufficient to ensure that failure of the intermediate system would be unlikely. | The proposed SFR-DC 75, 76, and 77 appear to elevate the importance of the intermediate coolant system as a safety grade system. With a non-radioactive coolant, maintaining the integrity of the intermediate coolant system boundary is an operational concern, not a safety issue. |
| 76 | *Fracture prevention of the intermediate coolant boundary.*<br>The intermediate coolant boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.<br><br>Rationale<br><br>This criterion is unique to the SFR design because, based on the information available to the staff, it is the only nuclear plant design for which there is an intermediate coolant loop. This criterion is identical to GDC 31 in 10 CFR 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed to avoid brittle and rapidly propagating facture modes. | The proposed new criteria 75, 76, and 77 should be removed. Quality of the intermediate coolant boundary within the context of a leakage that may prevent a structure, system, or component from performing any of its intended safety functions is already addressed in SFR-DC 70. More general considerations for sodium leakage detection, reaction prevention and mitigation are also captured in SFR-DC 73. If, for some designs, the intermediate coolant system is also utilized as a path for decay heat removal, then the quality of its boundary is covered in SFR-DC 34, 36, and 37.<br><br>The proposed SFR-DC 75, 76, and 77 appear to elevate the importance of the intermediate coolant system as a safety grade system. With a non-radioactive coolant, maintaining the integrity of the intermediate coolant system boundary is an operational concern, not a safety issue. |
| 77 | *Inspection of the intermediate coolant boundary.*<br>Components which are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the intermediate coolant boundary. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of coolant leakage.<br><br>Rationale | The proposed new criteria 75, 76, and 77 should be removed. Quality of the intermediate coolant boundary within the context of a leakage that may prevent a structure, system, or component from performing any of its intended safety functions is already addressed in SFR-DC 70. More general considerations for sodium leakage detection, reaction prevention and mitigation are also captured in SFR-DC 73. If, for some designs, the intermediate coolant system is also utilized as a path for decay heat removal, then the quality of its boundary is covered in SFR-DC 34, 36, and 37.<br><br>The proposed SFR-DC 75, 76, and 77 appear to elevate the importance of the intermediate coolant system as a safety grade system. With a non-radioactive |

| DOE Laboratory Team Comments on NRC SFR-DC<br>VII. Additional SFR-DC | | |
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| Criterion | **SFR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | This criterion is unique to the SFR design because, based on the information available to the staff, it is the only nuclear plant design for which there is an intermediate coolant loop. This criterion is identical to GDC 32 in 10 CFR 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed to avoid brittle and rapidly propagating facture modes. | coolant, maintaining the integrity of the intermediate coolant system boundary is an operational concern, not a safety issue. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>I. Overall Requirements | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 1 | Same as GDC<br><br>*Quality standards and records.*<br>Structures, systems, and components important to safety shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed. Where generally recognized codes and standards are used, they shall be identified and evaluated to determine their applicability, adequacy, and sufficiency and shall be supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program shall be established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained by or under the control of the nuclear power unit licensee throughout the life of the unit. | No comments. |
| 2 | Same as GDC<br><br>*Design bases for protection against natural phenomena.*<br>Structures, systems, and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components shall reflect: (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena and (3) the importance of the safety functions to be performed. | No comments. |
| 3 | Same as ARDC<br><br>*Fire protection.*<br>Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations ~~such as the~~ | On page 2 of the introductory material, NRC discusses Important-to-Safety (ITS), but it is not clear what the discussion implies. The proposed ARDC-3 language indicates that there is a difference between SSCs that are important to safety and SSCs that are safety related (SR) and potentially changes the applicability of the DC as stated in the first sentence. It is also noted that ITS is used throughout the GDC. SR is not used anywhere in the GDC.<br><br>The DOE December 2014 report (ML14353A246 and ML14353A248) provided a definition of the term “important to safety” to clarify and confirm its use, based |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>I. Overall Requirements | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | ~~containment and control room~~ with safety-related equipment or structures, systems, and ~~or~~ components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.<br><br>Rationale<br><br>The phrase containing examples where noncombustible and heat resistant materials must be used has been broadened to apply to all advanced reactor designs. | on industry's understanding of its intended meaning within the context of 10 CFR 50 Appendix A. This term was further addressed in DOE's response to NRC Question 40 (ML15204A579). The need for this clarification was also discussed in an NRC public meeting on January 21, 2015 (meeting summary available at ML15044A081), with DOE indicating that the ARDC's could be heavily impacted if NRC's understanding of the use of the term within Appendix A is different from that provided in the clarifying definition.<br><br>NRC should either confirm that "important to safety" means "safety related" within the context of Appendix A, or explain the difference between the two terms and provide a regulatory basis. |
| 4 | Same as ARDC<br><br>*Environmental and dynamic effects design bases.*<br>Structures, systems, and components important to safety shall be designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents~~, including loss of coolant accidents~~. These structures, systems, and components shall be appropriately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power unit. However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses reviewed and approved by the Commission demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.<br><br>Rationale<br><br>This change removes the LWR emphasis on loss of cooling accidents (LOCAs) that may not apply to some designs. For example, helium is not needed in a mHTGR to remove heat from the core during postulated accidents and does not have the same importance as water does to LWR designs to assure that fuel integrity is maintained. Therefore, a specific reference to "loss of coolant accidents" is not applicable to all designs. | It is not clear why the last sentence in the rationale for ARDC 4 has been inserted by the staff. For low pressure systems, it should be easy for designers and applicants to demonstrate that pipe whip phenomena do not apply, as implied in the last sentence of the criterion, which has not been modified for the ARDC. This last sentence in the rationale does not directly tie to any revision of this criterion, and it should be deleted. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>I. Overall Requirements | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design. Reference to pipe whip may not be applicable to designs that operate at low pressure. | |
| 5 | Same as GDC<br><br>*Sharing of structures, systems, and components.*<br>Structures, systems, and components important to safety shall not be shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety functions, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units. | NRC has not adopted the module language recommended by DOE for the modular HTGR version of Criterion 5. This leaves uncertainty regarding how to apply this criterion to plants that are designed and constructed in modular configurations. The language proposed by DOE, or something similar, addresses important aspects of sharing of SSCs that affect modular HTGR designs, and it should be incorporated into the modular HTGR version of Criterion 5. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>II. Multiple Barriers | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 10 | *Reactor design.*<br>The reactor ~~core~~ system and associated ~~coolant~~ heat removal, control, and protection systems shall be designed with appropriate margin to assure that specified acceptable ~~fuel~~ core radionuclide release design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.<br><br>Rationale<br><br>The specified acceptable fuel design limits (SAFDL), which prevents additional fuel failures during AOOs, has been replaced with the concept of specified acceptable radionuclide release design limits (SARRDL), which limits the amount of radionuclide inventory that escapes the fuel and circulates within the helium coolant boundary under normal operations and AOO conditions. The TRISO fuel of the mHTGR design is the primary fission product barrier and is expected to have very low incremental fission product release during AOOs. As noted in NUREG-1338, "Pre- application Safety Evaluation Report for the Modular High-Temperature Gas-Cooled Reactor (mHTGR)", and in the NRC staff's feedback on the next generation nuclear plant (NGNP) project white papers "Office of New Reactors Summary Feedback on Four Key Licensing Issues NGNP(ADAMS Package ML14174A626)," the TRISO fuel fission product transport and retention behavior under all expected operating conditions is the key to meeting dose limits as traditional defense in depth design features may not be included in a mHTGR. The SARRDL concept allows for some small increase in circulating radionuclide inventory during an AOO. To ensure the SARRDL is not violated during an AOO, a normal operation radionuclide inventory limit must also be established (i.e., appropriate margin). The radionuclide activity circulating within the helium coolant boundary is continuously monitored such that the normal operation limits and SARRDL are not exceeded.<br><br>The SARRDL will be established so that the most limiting license basis event does not exceed the siting regulatory dose limits criteria at the exclusion area boundary (EAB) and low population zone (LPZ), and also so that the 10 CFR 20.1301 annualized dose limits to the public are not exceeded at the EAB for normal operation and AOOs.<br>The concept of replacing SAFDL with SARRDL has not been reviewed or approved by the NRC. The concept of the TRSIO fuel being the primary | Change the first sentence of the first paragraph of the NRC rationale to correctly define SARRDL as the "specified acceptable **core** radionuclide release design limit". The sentence should also replace "circulates" with "circulates or condenses" to properly denote the contribution of both the circulating activity and the plateout activity to the SARRDL.<br><br>Rationale, First Paragraph, Third Sentence, states "…as traditional defense in depth design features may not be included in a mHTGR," but the text is not found in NUREG-1338 or in ADAMS Package ML14174A626. The staff's rationale wording could be taken to imply a deficiency where none exists. Regarding modular HTGRs, Page 3-7 of the 1989 version of NUREG-1338 states, "…designers have approached plant design and the means of maintaining defense-in-depth somewhat differently than the LWR designers." It would be more correct to characterize modular HTGR defense in depth by making the point from NUREG-1338 that different rather than traditional DID design features will be provided. Alternatively, the rationale text could be deleted in that it is unnecessary to support the rationale.<br><br>Rationale, Second Paragraph, Second Sentence, states, "The concept of replacing SAFDL with SARRDL has not been reviewed or approved by the NRC." The purpose of this sentence is unclear. The proposed SARRDL concept was submitted in the DOE report for NRC review and approval of the concept. The sentence does not contribute to or support the rationale. NRC separately requested public comment on the use of the SARRDL in its introduction to the draft design criteria. The sentence should be deleted from the rationale.<br><br>The Second Paragraph, Third Sentence of the Rationale: Replace "TRSIO" with "TRISO".<br><br>Third Paragraph, First Sentence of the Rationale: The helium purification system is not part of the "reactor system," and does not play a significant role in meeting the SARRDL. Its main purpose is to ensure ambient oxidant levels in helium are maintained within limits. Therefore, "such as the helium purification system," should be deleted from the 3rd paragraph of the rationale.<br><br>Rationale, Third Paragraph, Sentences 4 and 5 repeat information in Sentences 1-3 of the paragraph and should be deleted. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
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| | fission product barrier is intertwined the concept of a functional containment for mHTGR technologies. See the rationale for mHTGR-DC 16 for further information on the Commission's current position.<br><br>The word "core" has been replaced with "system" to include the components and internals of the mHTGR helium pressure boundary. Design features within the reactor system, such as the helium purification system, must be designed to assure that the SARRDLS are not exceeded during normal operations and AOOs. The word "coolant" has been replaced with "heat removal" as helium coolant inventory control for normal operation and AOOs is not necessary to meet the SARRDL due to the reactor system design. The word "core" has been replaced with "system" to denote that RCS design barriers exist for plate out and that systems such as the purification system contribute in meeting the specified acceptable core radionuclide release design limit (SARRDL). The word "coolant" has been replaced with "heat removal" as helium coolant inventory control for normal operation and AOOs is not necessary to meet the SARRDL due to the reactor system design. | |
| 11 | Same as ARDC<br><br>*Reactor inherent protection.*<br>The reactor core and associated ~~coolant~~ systems that contribute to reactivity feedback shall be designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.<br><br>Rationale<br><br>The wording has been changed to broaden the applicability from "coolant systems" to additional factors (including structures or other fluids) that may contribute to reactivity feedback. These systems are to be designed to compensate for rapid reactivity increase. | No comments. |
| 12 | *Suppression of reactor power oscillations.*<br>The reactor core and associated ~~coolant,~~ control, and protection systems shall be designed to assure that power oscillations which can result in conditions exceeding specified acceptable ~~fuel~~ core radionuclide release design limits are not possible or can be reliably and readily detected and suppressed. | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>II. Multiple Barriers | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | Rationale<br><br>Helium in the mHTGR does not affect reactor core susceptibility to coolant induced power oscillations; therefore, a separate mHTGR specific DC is appropriate. The word "coolant" was deleted and the SAFDLs were replaced by SARRDLs. The discussion regarding the SARRDL is given in mHTGR- DC 10. | |
| 13 | *Instrumentation and control.*<br>Instrumentation shall be provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those variables and systems that can affect the fission process, and the integrity of the reactor core, reactor helium ~~coolant~~ pressure boundary, and ~~reactor core, the reactor coolant pressure boundary, and the containment and its associated systems~~ functional containment.<br>Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges.<br><br>Rationale<br><br>"As appropriate" was removed to provide specificity to the criterion.<br><br>"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.<br><br>The criterion has been modified to reflect use of the modular HTGR functional containment. See mHTGR-DC 16 rationale. | Text was added to the criterion that assures "…the integrity of the reactor core, reactor helium pressure boundary, and functional containment." However, the reactor helium pressure boundary is an integral part of modular HTGR functional containment, so calling it out separate from the functional containment is redundant. Reference to the reactor helium pressure boundary should be deleted from the criterion to eliminate this redundancy. |
| 14 | *Reactor helium~~coolant~~ pressure boundary.*<br>The reactor helium~~coolant~~ pressure boundary shall be designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, and of unacceptable ingress of air, secondary coolant, or other fluids.<br><br>Rationale | The word "and" in the phrase "and of gross rupture" should be deleted. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>II. Multiple Barriers | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.<br><br>The addition of unacceptable air and fluid ingress, which is unique and critical to the mHTGR design, warranted the development of a mHTGR design specific criterion for the reactor helium pressure boundary. | |
| 15 | *Reactor helium pressure boundary ~~coolant system~~ design.*<br>The reactor helium pressure boundary ~~coolant system~~ and associated auxiliary, control, and protection systems shall be designed with sufficient margin to assure that the design conditions of the reactor helium pressure boundary ~~coolant pressure boundary~~ are not exceeded during any condition of normal operation, including anticipated operational occurrences.<br><br>Rationale<br><br>"Reactor coolant system" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs. | The NRC proposed criterion wording is redundant in stating, "The reactor helium pressure boundary...shall be designed with sufficient margin to assure that the design conditions of the reactor helium pressure boundary are not exceeded…" The criterion should be clarified by restating as: "The reactor system, vessel system, heat removal systems, and associated auxiliary, control, and protection systems shall be designed to assure that the design conditions of the reactor helium pressure boundary are not exceeded…" This change follows the structure of the original GDC and accurately identifies the modular HTGR systems that need to be designed to assure that the design conditions of the reactor helium pressure boundary are not exceeded. |
| 16 | *Containment design.*<br>A reactor functional containment, ~~and associated systems~~ consisting of ~~a structure surrounding the reactor and its cooling system or~~ multiple barriers internal and/or external to the reactor and its cooling system, shall be provided to ~~establish an essentially leak-tight barrier against the uncontrolled~~ control the release of radioactivity to the environment and to assure that the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.<br><br>Rationale<br><br>The term "functional containment" is applicable to advanced non-LWRs without a pressure retaining containment structure. mHTGR-DC 16 states that the functional containment:<br><br>"…shall be provided to control the release of radioactivity to the environment and to assure that the functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require." | Rationale, Last Paragraph, First Sentence: In addition to the GDCs listed as not applicable to the modular HTGR, GDCs 44, 45, 46, and 54 should be listed as not applicable. See other comments on those GDCs. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
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| | The DOE Report defines functional containment as: A barrier, or set of barriers taken together, that effectively limit the physical transport and release of radionuclides to the environment across a full range of normal operating conditions, anticipated operational occurrences, and accident conditions. Functional containment is relied upon to ensure that dose at the site boundary as a consequence of postulated accidents meets regulatory limits. Traditional containment structures also provide the reactor and SSCs important to safety inside the containment structure protection against accidents related to external hazards (turbine missiles, flooding, aircraft, etc.). Protection against accidents related to external hazards for mHTGRs is addressed in mHTGR-DCs 70-72.<br><br>The modular HTGR functional containment safety design objective is to meet 10 CFR 50.34, 52.79, 52.137, or 52.157 offsite dose requirements at the plant's exclusion area boundary (EAB) with margins. The DOE report further clarifies functional containment in section 7.1.4:<br><br>Modular HTGRs employ a functional containment that consists of an integrated set of five radionuclide retention barriers: 1) the coated fuel particle kernel, 2) the fuel particle coatings surrounding the particle kernel, 3) the carbonaceous matrix and graphite that surrounds the fuel particles, 4) the reactor helium pressure boundary, and 5) the reactor building.<br><br>NRC staff has brought the issue of functional containment to the Commission, and the Commission has found it generally acceptable as indicated in the SRMs to SECY-93-092 and SECY-03- 0047. NRC staff also provided feedback to the DOE on this issue as part of the Next Generation Nuclear Plant project. However, approval of the proposed approach to functional containment for the modular HTGR concept, with its emphasis on passive safety features and radionuclide retention within the fuel over a broad spectrum of off-normal conditions, would necessitate that the required fuel particle performance capabilities be demonstrated with a high degree of certainty. See the NRC staff's "Summary Feedback on Four Licensing Issues NGNP" regarding functional containment and fuel development and qualification (ML14174A774).<br><br>GDCs 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 55, 56, and 57 are not applicable to the mHTGR design since they address design criteria for | |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>II. Multiple Barriers | | |
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| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | pressure retaining containments in the traditional LWR sense. Requirements regarding the performance of the modular HTGR reactor building are addressed by new Criterion 71 (design basis) and Criterion 72 (provisions for periodic testing and inspection). | |
| 17 | *Electric power systems.*<br>An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable ~~fuel~~ core radionuclide release design limits and design conditions of the reactor helium~~coolant~~ pressure boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and functional containment integrity and other vital functions are maintained in the event of postulated accidents.<br><br>The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.<br><br>Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable ~~fuel~~ core radionuclide release design limits and design conditions of the reactor helium~~coolant~~ pressure boundary are not exceeded.<br>One of these circuits shall be designed to be available within a few seconds following a postulated ~~loss-of-coolant~~ accident to assure that core cooling, containment integrity, and other vital safety functions are maintained.<br><br>Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power | This criterion should maintain consistency with changes made in response to DOE Team comments on ARDC-17. Modifications to various other mHTGR-DC language will be needed should mHTGR-DC 17 be changed. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | from the transmission network, or the loss of power from the onsite electric power supplies.<br><br>Rationale<br><br>The requirements for offsite power are being retained for defense-in-depth considerations. This position was reinforced by a letter from the NRC to Dale Atkinson, Chief Operating Officer, NuScale Power, September 15, 2015 (ML15222A323). At the September 24, 2015 meeting of the Advisory Committee for Reactor Safeguards subcommittee on advanced reactor designs, this subject came up again and the subcommittee was supportive of keeping offsite power requirements in GDC 17 for the NuScale design.<br><br>“Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for mHTGRs.<br><br>The specified acceptable fuel design limits has been replaced with the specified acceptable core radionuclide release design limit. The discussion regarding the change to specified acceptable core radionuclide release design limit is given in GDC 10. | |
| 18 | Same as GDC<br><br>*Inspection and testing of electric power systems.*<br>Electric power systems important to safety shall be designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as onsite power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the offsite power system, and the onsite power system. | Revise as necessary to maintain consistency with changes made to ARDC 17 and 18. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | Rationale<br><br>GDC 18 is a design-independent companion criterion to GDC 17. | |
| 19 | Same as ARDC<br><br>*Control room.*<br>A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions~~including loss-of-coolant accidents~~. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem total effective dose equivalent ~~(~~TEDE) ~~whole body, or its equivalent to any part of the body~~, (TEDE) as defined in § 50.2 for the duration of the accident.<br><br>Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions.<br><br>Equipment at appropriate locations outside the control room shall be provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.<br><br>~~Applicants for and holders of construction permits and operating licenses under this part who apply on or after January 10, 1997, applicants for design approvals or certifications under part 52 of this chapter who apply on or after January 10, 1997, applicants for and holders of combined licenses or manufacturing licenses under part 52 of this chapter who do not reference a standard design approval or certification, or holders of operating licenses using an alternative source term under § 50.67, shall meet the requirements of this criterion, except that with regard to control room access and occupancy, adequate radiation protection shall be provided to ensure that radiation exposures shall not exceed 0.05 Sv~~ | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>II. Multiple Barriers | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | ~~(5 rem) total effective dose equivalent (TEDE) as defined in § 50.2 for the duration of the accident.~~<br><br>Rationale<br><br>The criterion was updated to remove specific emphasis on LOCA, which may be not appropriate for advanced designs such as the mHTGR.<br><br>Reference to “whole body, or its equivalent to any part of the body” has been updated to the current TEDE standard as defined in § 50.2.<br><br>Control room habitability requirement beyond that associated with radiation protection has been added to address concern that non-radionuclide accidents may also affect control room access and occupancy.<br><br>The last paragraph of the GDC has been eliminated for the ARDC because it is not applicable to future applicants. | |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>III. Reactivity Control | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 20 | *Protection system functions.*<br>The protection system shall be designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable ~~fuel~~ core radionuclide release design limit is not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety<br><br>Rationale<br><br>SAFDL has been replaced with SARRDL. The concept of using SARRDL is discussed for GDC 10. The quantitative value of the SARRDL will be design specific. The protection aspect of automatic operation and to protect normal operation and AOO limits and to sense accident conditions and initiate mitigating equipment has been preserved. | No comments. |
| 21 | Same as GDC<br><br>*Protection system reliability and testability.*<br>The protection system shall be designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that (1) no single failure results in loss of the protection function and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system shall be designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred. | No comments. |
| 22 | Same as GDC<br><br>*Protection system independence.*<br>The protection system shall be designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, shall be used to the extent practical to prevent loss of the protection function. | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>III. Reactivity Control | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 23 | Same as GDC<br><br>*Protection system failure modes.*<br>The protection system shall be designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air), or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced. | No comments. |
| 24 | Same as GDC<br><br>*Separation of protection and control systems.*<br>The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired. | No comments. |
| 25 | *Protection system requirements for reactivity control malfunctions.*<br>The protection system shall be designed to assure that specified acceptable ~~fuel~~ core radionuclide release design limits are not exceeded during any anticipated operational occurrence resulting from a ~~for~~ ~~any~~ single malfunction of the reactivity control systems. , ~~such as accidental withdrawal (not ejection or dropout) of control rods.~~<br><br>Rationale<br><br>Use ARDC except SAFDL is replaced with SARRDL. The concept of using SARRDLs is discussed for GDC 10. | No comments. |
| 26 | *Reactivity control system redundancy and capability.*<br>At least two independent reactivity control systems of different design principles shall be provided. One of the systems shall use control rods, preferably including a positive means for inserting the rods, and shall be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable ~~fuel~~ core radionuclide release design limits are not exceeded. The second reactivity control system shall be capable of | Next to Last Sentence of Criterion: “acceptable fuel design limits” should be changed to “acceptable core radionuclide release design limits.” This change would provide for consistent use of SARRDL in the modular HTGR criteria. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | reliably controlling the rate of reactivity changes resulting from planned, normal power changes ~~(including xenon burnout)~~ to assure acceptable fuel design limits are not exceeded. One of the systems shall be capable of holding the reactor core subcritical under cold conditions.<br><br>Rationale<br><br>Same rationale as the ARDC but with the additional revision of replacing specified acceptable fuel design limits with specified acceptable core radionuclide release design limits. The concept of using specified acceptable core radionuclide release design limits is discussed for GDC 10. | |
| 27 | Same as ARDC<br><br>*Combined reactivity control systems capability.*<br>The reactivity control systems shall be designed to have a combined capability~~, in conjunction with poison addition by the emergency core cooling system,~~ of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.<br><br>Rationale<br><br>None of the advanced non- LWR designs evaluated in the review utilized poison addition via an ECCS.<br><br>In addition, ARDC 34, *Residual heat removal*, combines the ECCS requirements in GDC 35 into ARDC 34, because none of the advanced non-LWR designs evaluated utilized an ECCS. Advanced non-LWR designs that do use poison addition or an ECCS will have to look to GDC 27 and GDC 35 for guidance. | No comments. |
| 28 | *Reactivity limits.*<br>The reactivity control systems shall be designed with appropriate limits on the potential amount and rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor helium~~coolant~~ pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor ~~pressure~~ vessel internals to impair significantly the capability to cool the core. ~~These postulated reactivity accidents shall include consideration of [rod ejection (unless prevented by positive means), rod~~ | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>III. Reactivity Control | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | ~~dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition].~~<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.<br><br>The list of "postulated reactivity accidents" has been deleted. Each design will have to determine its postulated reactivity accidents based on the specific design and associated risk evaluation. | |
| 29 | Same as GDC<br><br>*Protection against anticipated operational occurrences.*<br>The protection and reactivity control systems shall be designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences. | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>IV. Fluid Systems | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 30 | *Quality of reactor helium~~coolant~~ pressure boundary.*<br>Components which are part of the reactor helium ~~coolant~~ pressure boundary shall be designed, fabricated, erected, and tested to the highest quality standards practical. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor helium~~coolant~~ leakage.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs. | No comments. |
| 31 | *Fracture prevention of reactor helium ~~coolant~~ pressure boundary.*<br>The reactor helium ~~coolant~~ pressure boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.<br><br>Rationale<br><br>"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs. | No comments. |
| 32 | *Inspection of reactor helium~~coolant~~ pressure boundary.*<br>Components which are part of the reactor helium~~coolant~~ pressure boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the reactor ~~pressure~~ vessel.<br><br>Rationale | The words "…and leaktight…" should be deleted from criterion item (1). Consistent with NRC rationale for mHTGR-DC 33, no safety requirement is associated with reactor helium pressure boundary being leak tight. The cost of replacing helium inventory dictates the degree of leak tightness required. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>IV. Fluid Systems | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.<br><br>The staff modified the LWR GDC by replacing the term "reactor pressure vessel" with "reactor vessel", which staff believes is a more generically applicable term. | |
| 33 | Not applicable to modular HTGR.<br><br>Rationale<br><br>The mHTGR does not require reactor coolant inventory maintenance for small leaks to meet the SARRDLs, which replaces the concept of the SAFDLs as discussed in GDC 10. Therefore, ARDC 33 is not applicable to the mHTGR design. | No comments. |
| 34 | *Passive residual heat removal.*<br>A passive system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the ~~The~~ system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable ~~fuel~~ core radionuclide release design limits and the design conditions of the reactor helium~~coolant~~ pressure boundary are not exceeded.<br><br>During postulated accidents, the system safety function shall ~~be to~~ provide continuous effective cooling and to assure that the design conditions of the reactor helium pressure boundary are not exceeded.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>mHTGR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35. | Criterion paragraph 2 should maintain consistency with changes made in response to ARDC-34 comments. Electric power requirements contained in criterion paragraph 3 should maintain consistency with changes made in response to ARDC-17 comments.<br><br>Criterion Paragraph 2: "shall provide" should be changed to "shall be to provide" to provide wording consistent with the GDC. This comment also applies to the ARDC and SFR versions of the criterion.<br><br>The last paragraph of the NRC rationale addresses the role of the modular HTGR residual heat removal system in protecting passive heat removal geometry during postulated accidents. Since this function entails more than cooling just the core and reactor vessel, the first sentence of the rationale should be replaced with; "The ARDC "core cooling" was replaced with "cooling" to reflect the need to protect all components associated with the geometry for passive heat removal during postulated accidents." This change would improve technical accuracy of the rationale by capturing all components that need cooling. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | "Ultimate heat sink" has been added to clarify that if mHTGR-DC 44 is deemed not applicable to the design, the RHR system is then required to provide the heat removal path to the ultimate heat sink.<br><br>The word "passive" was added based on the definition of a modular HTGR. In definitions Section 3.1 of INL/EXT-14- 31179, the mHTGR design is defined as having passive heat removal due to a low power density.<br>"Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.<br><br>The specified acceptable core radionuclide release design limits replaces the ARDC specified acceptable fuel design limits as described in rationale to mHTGR-DC 10.<br><br>The ARDC "core cooling" was replaced with "cooling" in the second paragraph to reflect that the core and integrity of reactor vessel must be maintained by the residual heat removal system during postulated accidents. The last phrase was added to the second paragraph to assure that residual heat removal capability is sufficient to maintain the integrity of the reactor helium pressure boundary during postulated accidents. Maintaining the reactor helium pressure boundary is wording not currently in GDC 35 as the limiting postulated accident is a LOCA where primary coolant integrity is assumed lost. In advanced designs other accidents may be more limiting than a LOCA and hence the residual heat removal capability should be designed to ensure the reactor helium pressure boundary integrity is maintained. | |
| 35 | Same as ARDC<br><br>*Emergency core cooling.*<br>If the system as described in ARDC 34 does not provide continuous effective core cooling during postulated accidents and does not assure that the design conditions of the reactor coolant boundary are preserved; then a ~~A~~ system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant such that continuous effective core cooling is maintained.<br>Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities | The terms "abundant emergency core cooling" and "loss of reactor cooling" have no meaning for modular HTGRs. There are no modular HTGR accident scenarios that require abundant core cooling on an emergency basis. Modular HTGRs are relatively insensitive to undercooling events compared to LWRs. They can lose coolant pressure but they do not lose reactor coolant and their design basis includes the ability to handle coolant pressure loss with passive heat removal. There is no underlying safety basis that justifies retention of this criterion for modular HTGR technology. The criterion should be considered not applicable. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>In most advanced reactor designs, residual heat removal is addressed by ARDC 34. If the design is such that ARDC 34 is not adequate to ensure residual heat removal under normal operations and postulated accidents then additional system(s) are required and would be addressed by this ARDC 35 to ensure continuous effective core cooling. | |
| 36 | *Inspection of passive ~~emergency core cooling~~ residual heat removal system.*<br>The ~~emergency core cooling~~ ~~system~~ passive residual heat removal shall be designed to permit appropriate periodic inspection of important components~~, such as spray~~ ~~rings in the reactor pressure~~ ~~vessel, water injection~~ ~~nozzles, and piping,~~ to assure the integrity and capability of the system.<br><br>Rationale<br><br>The word "passive" was added based on the definition of a mHTGR. In definitions Section 3.1 of INL/EXT-14-31179, the mHTGR design is defined as having passive heat removal due to a low power density.<br><br>GDC 36 system is renamed and revised to provide for inspection of the residual heat removal systems as required for mHTGR-DC 34.<br><br>Deleted the example list as they apply to LWR designs and each specific design will have different important components associated with residual heat removal. | No comments. |
| 37 | *Testing of passive residual heat removal ~~emergency core~~ ~~cooling~~ system.*<br>The ~~emergency core cooling~~ passive residual heat removal system shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) the structural ~~and~~ ~~leaktight~~ integrity of its components, (2) the operability and performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the | Pursuant to the 3rd paragraph of the NRC rationale, Criterion Item (3) should be editorially revised to state, "… and (3) the operability of the system as a whole and, if applicable, under conditions as close to design as practical…"<br><br>The modular HTGR residual heat removal system is passive and does not rely on "…operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | performance of the full operational sequence that brings the system into operation, including operation of associated systems and interfaces with an ultimate heat sink and the transition from the active normal operation mode to the passive operation mode relied upon during postulated accidents including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.<br><br>Rationale<br><br>Criterion 37 has been renamed and revised for testing of the passive residual heat removal system required by modular HTGR-DC 34.<br><br>Section 2.3.4 of INL/EXT-10- 17997, "NGNP Mechanistic Source Terms White Paper, July 2010, ML102040260, notes the passive RCCS (using either air or water as heat transfer fluid) contributes to the modular HTGR safety basis and is subject to component integrity testing. However, Section 6.1 of INL/EXT-11-22708, "Modular HTGR Safety Basis and Approach", Aug 2011, ML11251A169, indicates that RCCS performance does not require "leaktight" conditions.<br><br>Some modular HTGR reactor cavity cooling system (RCCS) designs will provide continuous passive operation without need for a requirement to test the operation sequence that brings the system into operation; "if applicable" is included to recognize this contingency.<br><br>The criterion was modified to reflect the passive nature of the modular HTGR RCCS and the need to verify ability to transition the RCCS from active mode (if present) to passive mode during postulated accidents. | associated cooling water system." The words at the end of the criterion have no modular HTGR application and should be deleted pursuant to the last paragraph of the rationale. This comment makes the criterion more technically correct. |
| 38 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>IV. Fluid Systems | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 39 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 40 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 41 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 42 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 43 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>IV. Fluid Systems | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | |
| 44 | *Structural and equipment cooling. ~~Cooling water.~~*<br>In addition to the heat rejection capability of the passive residual heat removal system, ~~A~~ systems to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided, as ~~n~~ecessary. ~~The system safety function shall be~~ to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>Inserted "passive" based on system design for residual heat removal. If a specific mHTGR design can demonstrate that the reactor cavity cooling system (RCCS) provides indefinite core cooling capability, then structural and equipment cooling systems would not be needed. | If a modular HTGR design cannot demonstrate that the reactor cavity cooling system (RCCS) provides an indefinite core cooling capability, then the design is inadequate and needs to be redesigned to keep the facility consistent with the definition of a modular HTGR. GDCs 44, 45, and 46 address safety systems that are not present in a properly designed modular HTGR and are therefore not applicable on that basis. |
| 45 | Same as ARDC<br><br>*Inspection of structural and equipment cooling ~~water~~ systems.*<br>The ~~cooling water~~ structural and equipment cooling systems shall be designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the systems.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs. | See comment on modular HTGR Criterion 44. |
| 46 | Same as ARDC<br><br>*Testing of structural and equipment cooling ~~. water~~ systems.*<br>The structural and equipment cooling ~~water~~ systems shall be designed to permit appropriate periodic ~~pressure and~~ functional testing to assure (1) | See comment on modular HTGR Criterion 44. |

| Criterion | mHTGR-DC Language/ Rationale for Modification | Team Comments |
|---|---|---|
| | the structural ~~and leaktight~~ integrity of their ~~its~~ components, (2) the operability and the performance of the ~~active~~ system components ~~of the system~~, and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequences that bring~~s~~ the systems into operation for reactor shutdown and postulated accidents, including operation of associated systems. ~~and for loss-of-coolant accidents, including operation of~~ and applicable portions of the protection system and the transfer between normal and emergency power sources.<br><br>Rationale<br><br>This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs.<br><br>Specific mention of “pressure” testing has been removed yet remains a potential requirement should it be necessary as a component of “…appropriate periodic functional testing...” of cooling systems.<br><br>“Leaktight” integrity would be demonstrated through appropriate functional testing of system performance and operability.<br><br>“Active” has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature.<br><br>LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs. | |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>V. Reactor Containment | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 50 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 51 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 52 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 53 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 54 | Same as ARDC<br><br>Rationale<br><br>In that the specific design details of each mHTGR is unknown at this time, ARDC 54 should continue to apply to the mHTGR design. An applicant | NRC's proposed rationale for ARDC 54 indicates that it "only applies to designs employing containment structures". DOE has previously indicated, as a part of its GDC adaptation proposal that modular HTGR designs do not include or rely on a containment structure to provide radionuclide retention. The retention barriers provided to satisfy regulatory requirements for offsite dose instead include the fuel kernel, fuel particle coatings, matrix/graphite, and helium |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>V. Reactor Containment | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | could indicate in its application that its specific mHTGR design makes this GDC not applicable. | pressure boundary (as summarized on February 25, 2015 in a non-public seminar with the NRC staff). NRC has reflected this concept in mHTGR DCs 50-53 and 55-57. However, NRC's proposal that mHTGR-DC 54 be the "same as ARDC" is not consistent with either the rationale for ARDC 54, or with the rationales for mHTGR-DCs 50-53 and 55-57. The text for mHTGR-DC 54 should therefore be revised to "Not applicable to modular HTGR" to be consistent with the NRC's treatment of the functional containment topic in mHTGR-DCs.<br><br>It is also noted that the performance of the modular HTGR Reactor Building, which is not relied on to satisfy regulatory requirements for offsite dose, is addressed by new Criterion 71 (design basis) and Criterion 72 (provisions for testing and inspection). |
| 55 | Not applicable to modular HTGR.<br><br>Rationale<br><br>Lines that form a portion of the reactor coolant pressure boundary do not penetrate the reactor building. Therefore, this criterion does not apply. | No comments. |
| 56 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |
| 57 | Not applicable to modular HTGR.<br><br>Rationale<br><br>This criterion is not applicable to modular HTGR. Modular HTGRs designs do not have a "pressure retaining reactor containment structure", but instead rely on a multi-barrier functional containment configuration to control the release of radionuclides. See mHTGR-DC 16 rationale. | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>VI. Fuel and Radioactivity Control | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 60 | Same as GDC<br><br>*Control of releases of radioactive materials to the environment.*<br>The nuclear power unit design shall include means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid wastes produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity shall be provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment. | No comments. |
| 61 | Same as ARDC<br><br>*Fuel storage and handling and radioactivity control.*<br>The fuel storage and handling, radioactive waste, and other systems which may contain radioactivity shall be designed to assure adequate safety under normal and postulated accident conditions. These systems shall be designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage ~~coolant inventory~~ cooling under accident conditions.<br><br>Rationale<br><br>The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of non-LWR advanced reactors. However, some advanced designs may use dry fuel storage that incorporates cooling jackets that can be liquid-cooled or air-cooled to remove heat. This modification to this GDC allows for both liquid and air-cooling of the dry fuel storage containers. | No comments. |
| 62 | Same as GDC<br><br>*Prevention of criticality in fuel storage and handling.*<br>Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>VI. Fuel and Radioactivity Control | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | configurations. | |
| 63 | Same as GDC<br><br>*Monitoring fuel and waste storage.*<br>Appropriate systems shall be provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions. | No comments. |
| 64 | *Monitoring radioactivity releases.*<br>Means shall be provided for monitoring the reactor ~~containment~~ building atmosphere, ~~spaces containing components for recirculation of loss-of-coolant accident fluids,~~ effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.<br><br>Rationale<br><br>The underlying concept of monitoring radioactivity releases from the modular HTGR particle fuel to the reactor building, effluent discharge paths, and the plant environs applies. High radioactivity in the reactor building provides input to the plant protection system. In addition, the reactor building atmosphere is monitored for personnel protection. Recirculation of loss-of- coolant fluids (i.e., water) does not apply to the modular HTGR.<br><br>The descriptions of the associated atmospheres and spaces that are required to be monitored are revised to reflect the modular HTGR's different design configuration and functional containment arrangement. | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>VII. Additional mHTGR-DC | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| 70 | *Reactor vessel and reactor system structural design basis.*<br>The design of the reactor vessel and reactor system shall be such that their integrity is maintained during postulated accidents (1) to ensure the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and (2) to permit sufficient insertion of the neutron absorbers to provide for reactor shutdown.<br><br>Rationale<br><br>New modular HTGR design-specific GDC is necessary to assure reactor vessel and reactor system (including the fuel, reflector, control rods, core barrel, and structural supports) integrity is preserved for passive heat removal and for insertion of neutron absorbers. | No comments. |
| 71 | *Reactor building design basis.*<br>The design of the reactor building shall be such that during postulated accidents it structurally protects the geometry for passive removal of residual heat from the reactor core to the ultimate heat sink and provides a pathway for release of reactor helium from the building in the event of depressurization accidents.<br><br>Rationale<br><br>The reactor building functions are to protect and maintain passive cooling geometry and to provide a pathway for the release of helium from the building in the case of a line break in the reactor coolant pressure boundary. This newly established criterion assures that these safety functions are provided.<br><br>It is noted that the reactor building is not relied upon to meet the offsite dose requirements of 10 CFR 50.34 (10 CFR 52.79). | The reference to the “reactor coolant pressure boundary” in the first sentence of the Rationale should be changed to “reactor helium pressure boundary”. This change is needed to conform to standard terms used for mHTGRs, and to be consistent with other mHTGR design criteria (such as mHTGR-DCs 14, 15, 28, and 31). |
| 72 | *Provisions for periodic reactor building inspection*.<br>The reactor building shall be designed to permit (1) appropriate periodic inspection of all important structural areas and the depressurization pathway, and (2) an appropriate surveillance program.<br><br>Rationale<br><br>This newly established criterion regarding periodic inspection and surveillance provides assurance that the reactor building will perform its | No comments. |

| DOE Laboratory Team Comments on NRC Modular HTGR Design Criteria<br>VII. Additional mHTGR-DC | | |
|---|---|---|
| Criterion | **mHTGR-DC Language/<br>Rationale for Modification** | **Team Comments** |
| | safety functions of protecting and maintaining the configuration needed for passive cooling and providing a discharge pathway for helium depressurization events. | |

| | |
|---|---|
| **From:** | Kinsey, Jim C |
| **To:** | AdvancedRxDCComments Resource |
| **Cc:** | Holbrook, Mark; Wayne Moe; George Flanagan; Poore III, Willis P.; Randy Belles; Tanju Sofu; Tom King; Dave Alberstein; Fred Silady |
| **Subject:** | [External_Sender] Additional DOE Laboratory Team Comments |
| **Date:** | Wednesday, June 08, 2016 8:52:05 AM |
| **Attachments:** | Additional DOE Lab Team Comments.pdf |

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The DOE national laboratory team is providing the attached comments and feedback on the NRC's draft Advanced Reactor Design Criteria. These inputs are in addition to the comments previously transmitted on June 3, 2016.

We look forward to the continued engagement of the NRC staff, and to future industry stakeholder interactions through the Regulatory Guide public meeting and comment processes, as these draft design criteria are further updated and refined.

Please contact us if you have questions or require clariifyng information regarding these comments and observations.

Regards

Jim Kinsey

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Jim Kinsey
Regulatory Affairs Director
Idaho National Laboratory
(o) 208-526-6882
(m) 208-569-6751
jim.kinsey@inl.gov

**DOE lab team identification of areas needing further attention and dialogue during RG development**

1) From our general observations, and related dialogue with members of the advanced reactor stakeholder community, we believe that NRC should more completely describe the scope and key attributes of the initiative. The resulting additional clarity would promote more efficient and focused interactions with stakeholders going forward as the Regulatory Guide is developed. The key initiative attributes, established with DOE at the outset of the effort, include the following:

- The proposed advanced reactor design criteria seek to preserve the underlying safety principle(s) originally expressed in the corresponding GDC.
- Advanced reactor design criteria should recognize that safety design solutions can differ substantially from those associated with LWRs.
- The advanced reactor design criteria are intended to minimize the number of proposed GDC adaptations to those needed for improved regulatory certainty and clarity.
- As indicated by NRC, 10 CFR 50 has evolved over the years to address specific safety issues discovered as a result of operating experience and industry events (ATWS, aircraft impact, etc), although the results of those evolutions have not been added to the GDCs in Appendix A. This initiative retains that approach, and does not expand the adapted set of criteria to cover topics addressed elsewhere in 10 CFR 50.
- The advanced reactor design criteria are intended to apply to the six advanced reactor technology types identified in the DOE report, without a need to develop additional individual sets of technology-specific criteria prior to licensees establishing their individual principal design criteria. This point in particular is not clear to a number of stakeholders, many of whom seem to expect that there will be a specific set for each technology type.

2) The purpose of the initiative is to address the GDC to determine whether they apply to non-LWR designs and if not, to propose modifications to address the non-LWR design features. The DOE's national lab team concurs with the NRC's indication at the outset of the initiative that the Commission's 2008 policy on advanced reactors should be a central consideration in the development and subsequent review of the proposed GDC adaptations proposed in the advanced reactor design criteria. In some cases, this consideration appears to have been inconsistently applied within the proposed GDC adaptations, and therefore this joint initiative does not fully meet the objective of providing guidance that results in improved clarity for advanced reactor stakeholders. For example:

- NRC draft ARDC 27 regarding combined reactivity control systems considers the unique and updated approaches being pursued by advanced technology developers, and the proposed adaptation provides increased flexibility for satisfying the criterion, while still addressing the underlying safety basis of the GDC.
- In contrast, NRC draft ARDC 17 proposes to retain the existing and very prescriptive text associated with offsite and onsite power sources, even though that text is not consistent with the advanced reactor technology information that is available to support this initiative. This example indicates that in some cases, certain GDC are being treated as a set of minimum requirements, apparently in conflict with Appendix A language that the GDC are generally applicable to non-LWR technologies, and should be treated as guidance for establishing the associated PDCs. This example also appears to the DOE lab team to be inconsistent with the Commission's 2008 policy on advanced reactors.

| | |
|---|---|
| **From:** | Welling, Craig |
| **To:** | AdvancedRxDCComments Resource; Mazza, Jan; Jackson, Deborah; Jackson, Diane; Mayfield, Michael |
| **Cc:** | Sowinski, Thomas; O"Connor, Tom (NE-HQ); Welling, Craig; Golub, Sal; "Kinsey, Jim C" |
| **Subject:** | [External_Sender] Advanced Reactor Design Criteria comments from DOE |
| **Date:** | Monday, June 06, 2016 3:27:08 PM |
| **Attachments:** | Advanced Reactor Design Criteria comments by DOE 6 Jun 2016.docx |

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Mike,
In response to the 60 day informal comment request, attached are DOE high level comments on the draft GDCs posted on ADAMS.
Our laboratory team will send a separate more detailed input.
We welcome the opportunity to meet or have a call to discuss the comments.
Thank you again for your efforts on this important initiative.
Best regards,
Craig

From: Mayfield, Michael [mailto:Michael.Mayfield@nrc.gov]
Sent: Thursday, April 07, 2016 4:49 PM
To: Kelly, John E (NE) <JohnE.Kelly@Nuclear.Energy.Gov>; Golub, Sal <sal.golub@nuclear.energy.gov>; Welling, Craig <Craig.Welling@nuclear.energy.gov>
Subject: DRAFT GDC FOR INFORMAL COMMENT

This went to the public ADAMS site this afternoon at ML16096A420

You might not be able to see it until tomorrow but it is out for a 60 day informal comment period.

## DOE Comments on: NRC Invitation and Instructions

| Section | NRC Language | DOE Comments |
|---|---|---|
| **Process** | While developing the final RG, the NRC intends to consider the extent to which risk-informing the ARDC, SFR-DC, and mHTGR-DC is possible given the level of design information and data available. | This joint DOE-NRC initiative is intended to provide clarifying guidance for advanced (non-LWR) reactor developers, while minimizing the number of proposed adaptations to the existing GDC. Therefore, risk-informing the ARDC, SFR-DC, and modular HTGR-DC is outside the scope of the current initiative. |
| **Other Advanced Non-LWR Activities** | Paragraph 1: In addition to providing design criteria related to safety considerations, the staff is contemplating design considerations related to security requirements. This information is forthcoming and will be issued for comment separately. | Design considerations and associated regulatory requirements related to security are currently addressed outside of 10 CFR 50 Appendix A. When establishing this joint initiative, DOE and NRC agreed that its scope would be limited to the current content of Appendix A. Therefore, design considerations related to security should not be incorporated into the advanced reactor design criteria, although ODE recognizes the importance of considering security early in the reactor facility development process. |
| **Topics Open for Comment** | Should the current regulations that an applicant must address be incorporated into the ARDC? If so, which ones? | Appendix A of 10 CFR 50 does not contain a list of the current regulations that an applicant must address. Regulations that are not currently reflected in 10 CFR 50 Appendix A should similarly be excluded from the ARDC.<br><br>It is noted that this same question was raised during an NRC public meeting held on January 21, 2015 (meeting summary available at ML15044A081). The NRC and DOE responded that this was beyond the scope of this initiative, but could be a future endeavor. |
| | Are the ARDC generally applicable to the different types of non-LWRs being developed by different companies? | DOE issued a request to the advanced reactor stakeholder community at the outset of this joint initiative to gather current descriptions of the key features and available design detail for the technologies underdevelopment. This information was then used as a primary input to the development of the draft design criteria. That draft content was then further refined based on input DOE received in a public webinar (Sept. 2013), and two public workshops (April, 2014 and July, 2014) prior to submittal to the NRC. Based on these initiative related activities, DOE believes the ARDC to be generally applicable to the various types of non-LWRs under development. |
| | Is the approach to “functional containment” appropriately addressed in the proposed criteria? | DOE does not believe that “functional containment” is adequately addressed.<br><br>The ARDC language proposed by DOE in the December 2014 report (ML14353A246 and ML14353A248) was intended to be sufficiently general to allow the other reactor technology versions of Criterion 16 to be the same as the ARDC with no further technology specific modifications. The proposed language was chosen to encompass either a traditional containment structure that surrounds the reactor and its coolant system or a technology specific functional containment such as that used by the modular HTGR. Therefore, ARDC 16 should be revised in a manner consistent with the intent of the ARDC effort to maximize the suitability of the ARDC for various advanced non-LWR designs.. |

| Section | NRC Language | DOE Comments |
| --- | --- | --- |
| | | |

## DOE Comments on: Cross-cutting Topics

| Cross-cutting Topic | NRC Language | DOE Comments |
|---|---|---|
| Need to review and provide feedback on proposed definitions. | N/A | The DOE report provided a series of Definitions (report Section 3.1) that are intended to confirm a common understanding of the use of certain terms within the context of 10 CFR 50 Appendix A, and to provide added clarity regarding the use of selected terms unique to DOE's proposed non-LWRs design criteria. These definitions were reviewed with the NRC staff in a public meeting on January 21, 2015 (meeting summary available at ML15044A081), but were not reflected in the material provided for public comment. They should be addressed and confirmed in the pending Regulatory Guide.<br><br>In particular, NRC should either confirm that "important to safety" means "safety related" within the context of Appendix A, or explain the difference between the two terms and provide a regulatory basis. |

# ARDC-Specific DOE Comments

## I. Overall Requirements

| Criterion | ARDC Language/ Rationale for Modification | DOE Comments |
|---|---|---|
| 3 | *Fire protection.*<br>Structures, systems, and components important to safety shall be designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials shall be used wherever practical throughout the unit, particularly in locations ~~such as the~~ ~~containment and control room~~ with safety-related equipment or structures, systems, and ~~or~~ components important to safety. Fire detection and fighting systems of appropriate capacity and capability shall be provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Firefighting systems shall be designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.<br>Rationale<br>The phrase containing examples where noncombustible and heat resistant materials must be used has been broadened to apply to all advanced reactor designs. | As written, this text makes a distinction between "safety related equipment" and "structures, systems, and components important to safety" that is made no place else in the design criteria. Consistency with the GDC is needed with regard to the use of "important to safety" in the ARDC. Therefore, the following phrase should be removed to eliminate the confusion: "… safety-related equipment or…" |

## II. Multiple Barriers

| Criterion | ARDC Language/ Rationale for Modification | DOE Comments |
|---|---|---|
| 16 | Same as GDC<br><br>*Containment design.*<br>Reactor containment and associated systems shall be provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.<br><br>Rationale<br><br>For non-LWR technologies other than SFRs and mHTGRs, designers should use the current GDC to develop applicable principal design criteria. | ARDC 16 should be revised to reflect the DOE proposal.<br><br>The DOE-proposed ARDC 16 language, in conjunction with the definition of "functional containment" as provided in the DOE report, is sufficient to address radiological containment for a wide variety of advanced reactors without incurring technology-specific modifications. DOE's proposed criterion text was selected to encompass both traditional containment structures (and associated systems) and multi-barrier functional containment approaches and maintains consistency with Commission SRMs issued in response to SECY-93-0092, in response to SECY-03-0047, and with SECY-05-0006. |
| 17 | *Electric power systems.*<br>An onsite electric power system and an offsite electric power system shall be provided to permit functioning of structures, systems, and components important to safety. The safety function for each system (assuming the other system is not functioning) shall be to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant ~~pressure~~ boundary are not exceeded as a result of anticipated operational occurrences and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.<br><br>The onsite electric power supplies, including the batteries, and the onsite electric distribution system, shall have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.<br><br>Electric power from the transmission network to the onsite electric distribution system shall be supplied by two physically independent circuits (not necessarily on separate rights of way) designed and located so as to minimize to the extent practical the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits shall be designed to be available in sufficient time following a loss of all onsite alternating current power supplies and the other offsite electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the reactor coolant ~~pressure~~ boundary are not exceeded. One of these circuits shall be designed to be available within a few seconds following a postulated ~~loss-of-~~ ~~coolant~~ accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. | ARDC 17 should be revised to reflect the DOE proposal.<br><br>The Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" (ML082750370) summarizes its expectation that advanced reactors will provide enhanced margins of safety and/or use simplified, inherent, passive, or other innovative means to accomplish their safety and security functions. NRC's proposed content for ARDC 17 appears to be in conflict with this Commission expectation since the offsite power requirements typically associated with plants that rely on active safety systems are retained. This apparently conflicting ARDC content may discourage advanced reactor designers from pursuing the Commission's policy expectations regarding reduced reliance on active safety systems. |

| Criterion | ARDC Language/ Rationale for Modification | DOE Comments |
|---|---|---|
| | Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit, the loss of power from the transmission network, or the loss of power from the onsite electric power supplies.<br><br>Rationale<br><br>The requirements for offsite power are being retained for defense-in-depth considerations. This position was reinforced by a letter from the NRC to Dale Atkinson, Chief Operating Officer, NuScale Power, September 15, 2015 (ML15222A323). At the September 24, 2015 meeting of the Advisory Committee for Reactor Safeguards subcommittee on advanced reactor designs, this subject came up again and the subcommittee was supportive of keeping offsite power requirements in GDC 17 for the NuScale design.<br><br>LWR emphasis on LOCAs may not apply to non-LWR designs. For example, helium is not needed in an HTGR to remove heat from the core during postulated accidents and does not have the same importance as water does to LWR designs to assure that fuel integrity is maintained. LOCAs may still require analysis in conjunction with postulated accidents if relevant to the design.<br><br>Reactor coolant pressure boundary has been relabeled as “reactor coolant boundary” to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non- LWRs that operate at either low or high pressure. | |

| Criterion | ARDC Language/ Rationale for Modification | DOE Comments |
|---|---|---|
| 34 | *Residual heat removal.*<br>A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the ~~The~~ system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant ~~pressure~~ boundary are not exceeded.<br><br>During postulated accidents, the system safety function shall provide continuous effective core cooling and to assure that the design conditions of the reactor coolant boundary are not exceeded.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>ARDC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35.<br><br>"Ultimate heat sink" has been added to clarify that if ARDC 44 is deemed not applicable to the design, the RHR system is then required to provide the heat removal path to the ultimate heat sink.<br><br>Reactor coolant pressure boundary has been relabeled as "reactor coolant boundary" to create a more broadly applicable non-LWR term that defines the boundary without giving any implication of system operating pressure. As such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.<br><br>Text of first paragraph has been amended and the second paragraph added to clarify requirements that are applicable following normal operation including AOOs, and during postulated accidents following the precedent of NUREG-1368, "Pre-application SER for PRISM LMR."<br><br>The last phrase was added to the second paragraph to assure that residual heat removal capability is sufficient to maintain the integrity of the reactor coolant boundary during postulated accidents. Maintaining the reactor coolant boundary is wording not currently in GDC 35 as the limiting postulated accident is a LOCA where primary coolant integrity is assumed lost. In advanced designs other | DOE has previously reviewed the content and structure of the existing General Design Criteria included in Appendix A of 10 CFR 50, and has determined that various criteria (such as GDC's 4, 14, and 31) address the aspects of reactor coolant boundary integrity during postulated accidents. Therefore, the second paragraph of ARDC 34 should be revised to "During postulated accidents, the system safety function shall provide continuous effective core cooling." |

| Criterion | ARDC Language/<br>Rationale for Modification | DOE Comments |
|---|---|---|
| | accidents may be more limiting than a LOCA and hence the residual heat removal capability should be designed to ensure the reactor coolant boundary integrity is maintained.<br><br>The third paragraph addresses RHR system redundancy. ARDC 17 requires reliable power systems for SSCs performing vital safety functions and must be of adequate capacity and capability to operate during postulated accidents. There may be various combinations of power supply employed to address power reliability. | |
| 35 | *Emergency core cooling.* If the system as described in ARDC 34 does not provide continuous effective core cooling during postulated accidents and does not assure that the design conditions of the reactor coolant boundary are preserved; then a ~~A~~ system to provide abundant emergency core cooling shall be provided. The system safety function shall be to transfer heat from the reactor core following any loss of reactor coolant such that continuous effective core cooling is maintained. Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>In most advanced reactor designs, residual heat removal is addressed by ARDC 34. If the design is such that ARDC 34 is not adequate to ensure residual heat removal under normal operations and postulated accidents then additional system(s) are required and would be addressed by this ARDC 35 to ensure continuous effective core cooling. | ARDC 35 content should be deleted, consistent with the DOE proposal.<br><br>DOE did not propose ARDC 35 content, but instead indicated that; "If a separate ECCS system is required for an advanced reactor, the PDC process for that reactor must look directly to GDC 35 for guidance." A “contingency” system should not be assumed as an arbitrary design option. |
| 38 | *Containment heat removal.*<br>A system to remove heat from the reactor containment shall be provided as necessary ~~The system safety function shall be~~ to maintain ~~reduce rapidly, consistent with the functioning of other associated systems,~~ the containment pressure and temperature within acceptable limits following ~~following any loss-of-coolant~~ postulated accidents. ~~and maintain them at acceptably low levels.~~<br><br>Suitable redundancy in components and features, including electric power systems, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety | The basis for adding “…including electric power systems” to criteria 38 and 41 is unclear and not addressed in the rationale. Electric power systems are included under components and features, thereby making the new text unnecessary. |

| Criterion | ARDC Language/<br>Rationale for Modification | DOE Comments |
|---|---|---|
| | function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>"…as necessary…" is meant to condition ARDC requiring heat removal for conventional containments which are found to require heat removal measures.<br><br>LOCA reference has been removed to provide for any postulated accident that might affect the containment structure.<br><br>Containment structure safety system redundancy is addressed in second paragraph. | |

# SFR-Specific DOE Comments

| Criterion | SFR-DC Language/<br>Rationale for Modification | DOE Comments |
|---|---|---|
| 16 | *Containment design.*<br>A reactor containment consisting of a high strength, low leakage, pressure retaining structure surrounding the reactor and ~~associated~~ its cooling systems, shall be provided to ~~establish an essentially leak-tight barrier against the uncontrolled~~ control the release of radioactivity to the environment and to assure that the reactor containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.<br><br>The containment leakage shall be restricted to be less than that needed to meet the acceptable onsite and offsite dose consequence limits as specified in 10 CFR Part 50.34 for postulated accidents.<br><br>Rationale<br><br>The Commission approved the staff's recommendation to restrict the leakage of the containment to be less than that needed to meet the acceptable onsite and offsite dose consequence limits [Ref. SRM, SECY-93-092].<br>Therefore, the Commission agreed that the containment leakage for advanced reactors, similar to and including PRISM, should not be required to meet the "essentially leaktight" statement in GDC16. [Ref: NUREG-1368].<br><br>Also, ARDCs and SFR-DCs 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57 in the DOE report refer to containment in the traditional sense in that these SFR-DCs specify traditional containment systems design, inspection, and testing (including leakage rate testing).<br><br>Furthermore, all past, current, and planned SFR designs use a high strength, low leakage, pressure retaining containment concept which aims to provide a barrier to contain the fission products and other substances and to control the release of radioactivity to the environment. | Refer to comments on ARDC 16.<br><br>SFR-DC comment: Change first sentence wording “its cooling systems” to “its primary cooling system”<br><br>(SFR containment designs surround only the primary cooling system. There is no need to include the intermediate loop within the containment since this system will not contain radioactive materials.) |
| 34 | *Residual heat removal.*<br>A system to remove residual heat shall be provided. For normal operations and anticipated operational occurrences, the ~~The~~ system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core to an ultimate heat sink at a rate such that specified acceptable fuel design limits and the design conditions of the ~~reactor~~ primary coolant boundary are not exceeded.<br><br>During postulated accidents, the system safety function shall transfer heat from the | During postulated accidents, localized boiling may occur, and is acceptable, under the definition of continued effective core cooling.<br>Therefore, the phrase “sodium boiling is precluded” should be removed. |

| Criterion | SFR-DC Language/<br>Rationale for Modification | DOE Comments |
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| | reactor core at a rate such that fuel and clad damage that could interfere with continued effective cooling is prevented, sodium boiling is precluded, and the design conditions of the primary coolant boundary are not exceeded.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>A passive boundary shall separate primary coolant from the working fluid of the residual heat removal system and any fluid in the residual heat removal system that is separated from the primary coolant by a single passive barrier shall not be chemically reactive with the primary coolant. In addition, the working fluid of residual heat removal system shall be at a higher pressure than the primary coolant system.<br><br>Rationale<br><br>SFR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35.<br><br>“Ultimate heat sink” has been added to clarify that if SFR-DC 44 is deemed not applicable to the design, the RHR system is then required to provide the heat removal path to the ultimate heat sink.<br><br>“Reactor coolant pressure boundary” has been relabeled as “primary coolant boundary” to reflect that the SFR primary system operates at low- pressure and to conform to standard terms used in the LMR industry. The use of the term “primary” indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system.<br><br>The second paragraph was added to clarify that the safety function of the residual heat removal system during postulated accidents is to provide continuous effective core cooling. For SFRs, that cooling is provided at a rate sufficient to prevent propagation of fuel failures. The last phrase was added to the paragraph to assure that residual heat removal capability is sufficient to maintain the integrity of the primary coolant boundary during postulated accidents.<br><br>A paragraph from NUREG- 1368 (page 3-41) was added describing the characteristics of the residual heat removal working fluid and its associated operating pressure. A single passive barrier is adequate defense in depth when the residual heat removal working fluid is not chemically reactive with the primary coolant. If chemically reactive at least | In the fourth paragraph, the phrase “…shall not be chemically reactive with the primary coolant…” should be replaced with the phrase “…shall be chemically compatible with the primary coolant…”<br><br>[Requiring that the RHRS coolant shall be chemically nonreactive with sodium can be open to misinterpretation. Instead, requiring that RHRS coolant to be chemically compatible with sodium is more appropriate. For example, NaK (a possible RHRS coolant) is chemically compatible with sodium; however, it is not chemically nonreactive with sodium.] |

| Criterion | SFR-DC Language/<br>Rationale for Modification | DOE Comments |
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| | two passive barriers must separate the two systems. The higher pressure requirement is to ensure any leakage in the interface between the two systems does not result in a release of radioactive primary coolant to the non-radioactive part of the heat transport system. | |
| 75 | *Quality of the intermediate coolant boundary.*<br>Components which are part of the intermediate coolant boundary shall be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety functions to be performed.<br><br>Rationale<br><br>This criterion is unique to the SFR design because, based on the information available to the staff, it is the only nuclear plant design for which there is an intermediate coolant loop. This criterion is identical to GDC 30 in 10 CFR 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed , fabricated, and tested using quality standards and controls sufficient to ensure that failure of the intermediate system would be unlikely. | The proposed new criteria 75, 76, and 77 should be removed. Quality of the intermediate coolant boundary within the context of a leakage that may prevent a structure, system, or component from performing any of its intended safety functions is already addressed in SFR-DC 70. More general considerations for sodium leakage detection, reaction prevention and mitigation are also captured in SFR-DC 73. If, for some designs, the intermediate coolant system is also utilized as a path for decay heat removal, then the quality of its boundary is covered in SFR-DC 34, 36, and 37. |
| 76 | *Fracture prevention of the intermediate coolant boundary.* The intermediate coolant boundary shall be designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design shall reflect consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing, and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.<br><br>Rationale<br><br>This criterion is unique to the SFR design because, based on the information available to the staff, it is the only nuclear plant design for which there is an intermediate coolant loop. This criterion is identical to GDC 31 in 10 CFR 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed to avoid brittle and rapidly propagating facture modes. | The proposed new criteria 75, 76, and 77 should be removed. Quality of the intermediate coolant boundary within the context of a leakage that may prevent a structure, system, or component from performing any of its intended safety functions is already addressed in SFR-DC 70. More general considerations for sodium leakage detection, reaction prevention and mitigation are also captured in SFR-DC 73. If, for some designs, the intermediate coolant system is also utilized as a path for decay heat removal, then the quality of its boundary is covered in SFR-DC 34, 36, and 37. |
| 77 | *Inspection of the intermediate coolant boundary.*<br>Components which are part of the intermediate coolant boundary shall be designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leaktight integrity, and (2) an appropriate material surveillance program for the intermediate coolant boundary. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of coolant leakage. | The proposed new criteria 75, 76, and 77 should be removed. Quality of the intermediate coolant boundary within the context of a leakage that may prevent a structure, system, or component from performing any of its intended safety functions is already addressed in SFR-DC 70. More general considerations for sodium leakage detection, reaction prevention and mitigation are also captured in SFR-DC 73. If, for some designs, the intermediate coolant system is also |

| Criterion | SFR-DC Language/<br>Rationale for Modification | DOE Comments |
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| | Rationale<br><br>This criterion is unique to the SFR design because, based on the information available to the staff, it is the only nuclear plant design for which there is an intermediate coolant loop. This criterion is identical to GDC 32 in 10 CFR 50, Appendix A, and is intended to ensure that, similar to the reactor coolant pressure boundary, the intermediate coolant boundary is designed to avoid brittle and rapidly propagating facture modes. | utilized as a path for decay heat removal, then the quality of its boundary is covered in SFR-DC 34, 36, and 37. |

## Modular HTGR-Specific DOE Comments

| Criterion | mHTGR-DC Language/<br>Rationale for Modification | DOE Comments |
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| 44 | Structural and equipment cooling. ~~Cooling water.~~<br>In addition to the heat rejection capability of the passive residual heat removal system, ~~A~~ systems to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided, as necessary. ~~The system safety function shall be~~ to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.<br><br>Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities shall be provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.<br><br>Rationale<br><br>Inserted "passive" based on system design for residual heat removal. If a specific mHTGR design can demonstrate that the reactor cavity cooling system (RCCS) provides indefinite core cooling capability, then structural and equipment cooling systems would not be needed. | There is no technical basis for applying GDCs 44, 45, and 46 to the modular HTGR. This criterion should, therefore, be revised to "Not applicable to modular HTGR" as proposed by DOE.<br><br>(If a modular HTGR design cannot demonstrate that the reactor cavity cooling system (RCCS) provides an indefinite core cooling capability, then the design is inadequate and needs to be redesigned to keep the facility consistent with the definition of a modular HTGR.) |