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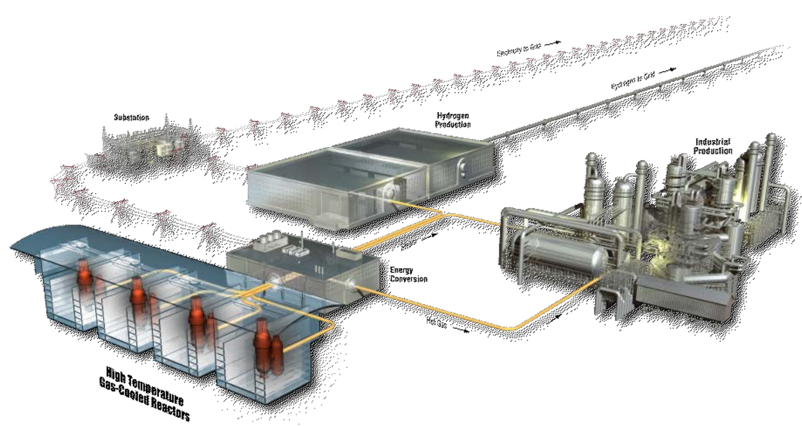
Revision 0

Summary of the Advanced Reactor Design Criteria (ARDC) Phase 2 Activities

Mark R. Holbrook

September 2015



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SUMMARY

This report provides a year-end summary reflecting the progress and status of development of proposed regulatory design criteria for advanced non-light water reactor designs in accordance with the Level 3 milestone M3AT-15IN2001017 in work package AT-15IN200101. These criteria have been designated as Advanced Reactor Design Criteria (ARDC), and they provide guidance to future applicants for addressing the General Design Criteria (GDC) that are currently applied specifically to light water reactor (LWR) designs. The report provides a summary of activities related to the various tasks associated with Nuclear Regulatory Commission (NRC) development of regulatory guidance for Sodium Fast Reactor (SFR) and modular High Temperature Gas-cooled Reactor (modular HTGR) designs.

The report summarizes activities associated with Phase 2 of ARDC and regulatory guidance development tasks. Phase 2 of this effort is currently in progress under the leadership of the NRC.

CONTENTS

FIGURES

ACRONYMS

ARDC	Advanced Reactor Design Criteria
ANL	Argonne National Laboratory
DOE	Department of Energy
GDC	General Design Criteria
HTGR	High Temperature Gas-cooled Reactor
INL	Idaho National Laboratory
LWR	light water reactor
NRC	Nuclear Regulatory Commission
ORNL	Oak Ridge National Laboratory
PDC	principal design criteria
SFR	Sodium Fast Reactor

Summary of the Advanced Reactor Design Criteria (ARDC) Phase 2 Activities

1. Purpose

This report provides an end-of-year summary reflecting the progress and status of proposed regulatory design criteria for advanced non-LWR designs in accordance with the Level 3 milestone in M3AT-15IN2001017 in work package AT-15IN200101. These criteria have been designated as ARDC, and they provide guidance to future applicants for addressing the GDC that are currently applied specifically to LWR designs. The report provides a summary of Phase 2 activities related to the various tasks associated with ARDC development and the subsequent development of example adaptations of ARDC for Sodium Fast Reactor (SFR) and modular High Temperature Gas-cooled Reactor (HTGR) designs.

2. Background

Nuclear Regulatory Commission (NRC) requirements for reactor licensing and deployment include the requirement in 10 CFR 50.34 to establish Principal Design Criteria (PDC) derived from the General Design Criteria (GDC) of 10 CFR 50, Appendix A. Since the GDC in Appendix A were created primarily for Light Water Reactors (LWRs), this requirement becomes challenging for future license applicants pursuing advanced (non-LWR) reactor technologies and designs.

During 2012, the Department of Energy (DOE) initiated a Technical Review Panel process to evaluate certain advanced reactor concepts for viable commercial deployment. Early in that process, Technical Review Panel members and advanced reactor designers voiced a need to develop a compatible regulatory framework for advanced non-LWRs to reduce risks and uncertainty to the advanced reactor industry. In addition, the NRC provided "Report to Congress: Advanced Reactor Licensing," dated August 2012, that noted several prospective advanced reactor vendors who identified a need for refined regulatory guidance pertaining specifically to their advanced non-LWR designs.

To support this need, DOE and NRC considered approaches for establishing a regulatory framework for advanced non-LWRs. From this, it was agreed that supporting a joint initiative for the development of Advanced Reactor Design Criteria (ARDC) for use by advanced reactor designers and future license applicants would be an important first step in developing that framework.

3. Objective

The objective of the ARDC development activity is to support the NRC in development of regulatory guidance that can be used as guidance by advanced reactor designers and future applicants to establish PDC for advanced non-LWR designs.

4. Scope

The Advanced Reactor Regulatory Framework Development work scope has been developed in two phases; Phase 1 was performed primarily by a DOE and national laboratory team, and involved development of the proposed set of ARDC, including additional development of design-specific criteria.

Phase 2 is being performed primarily by NRC and includes the initiation of their regulatory development process, followed by the issuance of regulatory guidance to the industry.

The ARDC development activity considered design attributes and regulatory needs concerning the following advanced reactor technologies: SFRs, Lead Fast Reactors, Gas-cooled Fast Reactors, modular HTGRs, Fluoride High-Temperature Reactors, and Molten Salt Reactors.

5. Completion of Phase 1 Tasks

The results of the ARDC team's Phase 1 analysis are contained in INL/EXT-14-31179, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors," dated December 2014. This report finalized the proposed ARDC language and documented the results of generic and technology-specific design criteria development reflecting the resolution of stakeholder comments and outstanding issues and was developed under PLN-2690, "Idaho National Laboratory Advanced Reactor Technologies Technology Development Office Quality Assurance Plan," Rev. 12, dated February 2, 2014.

The proposed ARDC provide specific inputs and recommendations to support the NRC staff's issuance of guidance reflecting how developers of the selected advanced reactor technology types could adapt the existing GDC contained in 10 CFR 50, Appendix A, to the development of their respective PDC while retaining the underlying safety principles of the GDC. The relationship among the 10 CFR 50 GDC, the ARDC, the two sets of technology-specific design criteria contained in the report, and the PDC that a future license applicant is required to submit is reflected in Figure 1.

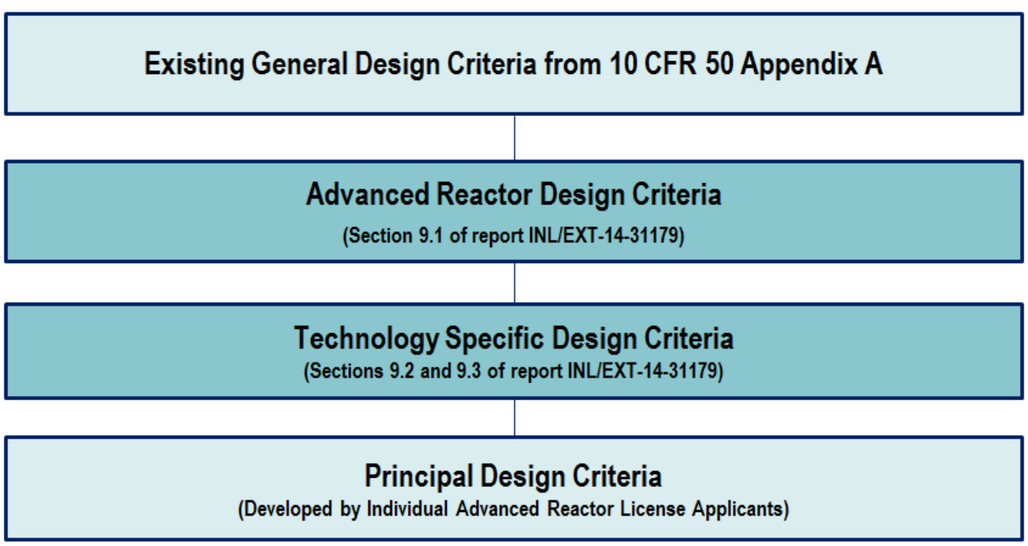

Figure 1. Design criteria relationship.

Section 9 of the report includes the proposed ARDC, SFR-specific design criteria, modular HTGR-specific design criteria, and a special table (found in Section 9.4) that compares all of the modified versions of design criteria language to the original GDC.

6. Summary of Fiscal Year 2015 ARDC Phase 2 Activities

As noted in Section 4 of this report, Phase 2 is being performed primarily by NRC and includes their regulatory development process and the subsequent issuance of regulatory guidance for development of principal design criteria for advanced (non-LWR) designs. The ARDC team has been (and will continue to be) available to respond to any questions related to the proposed ARDC (and design-specific criteria) that were documented in report INL/EXT-14-31179, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors."

On January 21, 2015, DOE and ARDC team met with the NRC staff in a public meeting to initiate discussions of the contents of DOE's ARDC report. During this meeting, the NRC staff provided an overview of the DOE-NRC advanced reactor licensing strategy initiative. In addition, the ARDC team provided a series of presentations that addressed the ARDC development process and the content found in report INL/EXT-14-31179, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors."

To support development of responses to anticipated NRC staff questions, a desktop guide was developed in May 2015 to define the process for developing timely responses to the NRC questions. This guide described the steps necessary to process responses to NRC questions within a predetermined schedule and assigned responsibilities for accomplishment of those steps. NRC question tracking and status tools also were developed in May.

On June 5, 2015, the NRC transmitted a series of staff questions related to their review of DOE report INL/EXT-14-31179, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors." This transmittal included 40 specific staff questions related to the development of proposed advanced reactor design criteria that were described by the DOE report. The ARDC team developed responses for each of the NRC staff questions, and conducted internal and external reviews of the response material. The final material was transmitted by DOE to the NRC on July 15, 2015. This final material is included in Appendix A of this report.

On August 17, 2015, the NRC transmitted three additional questions related to the NRC staff's review of the DOE ARDC report cited above. On August 20, 2015, the ARDC team held a meeting to discuss the content of the new questions and individual question response assignments were made. Completion of these responses is in progress at the time of this report development. The content of these additional NRC questions is included in Appendix B of this report.

7. Future ARDC Activities

Phase 2 of the initiative is being managed by the NRC and is expected to involve review of the Phase 1 work products and issuance of regulatory guidance resulting from the review. This process will include resolution of outstanding NRC staff technical questions and comments gathered through the public interaction process. The ARDC will remain available to assist the NRC with reviews of draft regulatory guidance as requested during Phase 2. NRC has stated that they intend to develop and issue regulatory guidance commensurate with an official NRC staff position with a completion target of the end of calendar year 2016.

Appendix A

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” - NRC Project # 0814

Appendix A

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" - NRC Project # 0814

NRC Staff Questions on the DOE Report:

Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors
Department of Energy – Idaho National Laboratory

Docket No. PROJ 0814

1. ARDC 16

a. The Department of Energy (DOE) Report defines and introduces functional containment as: "*A reactor functional containment, 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...*"

Advanced reactor design criteria (ARDC) 16, which applies to sodium fast reactor design criteria (SFR-DC) 16 and modular high temperature reactor design criteria (mHTGR-DC) 16, adopt the definition of functional containment which removes the "essentially leak tight" qualification. ARDCs and SFR-DCs 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57 refer to containment in the traditional sense in that these ARDCs/SFR-DCs specify traditional containment systems design, inspection, and testing (including leakage rate testing). The mHTGR-DCs assert that general design criteria (GDCs) 38, 39, 40, 41, 42, 43, 50, 51, 52, 53, 54, 55, 56, and 57 are not applicable to the mHTGR design. The mHTGR design features a vented low pressure reactor building that does not necessarily retain radionuclides. Was applying the functional containment concept only to mHTGRs considered?

b. ARDC 16 further states that the functional containment "*…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.*" For mHTGRs, the vented low pressure reactor building is designed to allow the controlled release of radionuclides, but other nonlight-water reactor (non-LWR) technologies may employ traditional essentially leak-tight containments. Did the DOE intend for these technologies to allow a controlled release from an essentially leak-tight containment? Should a new ARDC/SFR-DC be developed to address structures, systems, and components (SSCs) that control the release of radionuclides and ensure that Title 10 of the *Code of Federal Regulations* (10 CFR) Part 20 limits to workers and the public are not exceeded for postulated accidents?

DOE Response:

While ARDC 16 introduces and discusses the application of functional containment, the criterion does not define that term. As stated in Section 3.1 of the DOE report, functional containment is defined 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.*

The following responses to the two-part staff question are provided with this definition specifically in mind:

a. Application of the “functional containment” concept only to modular HTGRs was considered. However, that approach constituted an exception to the criteria requirements in Appendix A rather than an expansion to different advanced reactor types. Because ARDCs are intended to provide the broadest possible design guidance while minimizing the need for exceptions, they were developed to be as inclusive as possible and generally applicable to the six advanced reactor technology types summarized in Section 1.3 of the DOE report.

   A definition of “functional containment” was established that is sufficiently broad to encompass both a high-pressure, low-leakage containment structure and the multi-barrier design approach of a modular HTGR. Using this definition, ARDC 16 requires that barrier(s) and associated systems be provided to assure adequate control of radioactivity releases to the environment. This is done without dictating a specific design solution or relying on prescriptive terminology like “an essentially leak-tight barrier” that infers preference for a containment structure. Instead, the criterion focuses on assuring a reliable and adequate radionuclide containment function is provided. The emphasis on overall performance rather than traditional design feature allows for fuller consideration of new approaches as is represented by the multi-barrier design.

   This approach is consistent with positions taken by the NRC staff in SECY-93-092, “Issues Pertaining to the Advanced Reactor (PRISM, MHTGR, and PIUS) and CANDU 3 Designs and Their Relationship to Current Regulatory Requirements”. In that document, the staff proposed to utilize a standard based upon containment functional performance to evaluate the acceptability of proposed designs rather than rely exclusively on prescriptive containment design criteria. The staff also proposed a containment performance criterion in which designs must be adequate to meet the onsite and offsite radionuclide release limits for the event categories within their design envelope.

   In the Staff Requirements Memorandum (SRM) to SECY-93-092, the Commission indicated that a conventional LWR leak-tight containment should not be required for advanced reactor designs. It approved the use of containment functional design criteria for evaluating the acceptability of proposed containment designs rather than the use of prescriptive design criteria. This position regarding containment allows the acceptance of containments with leak rates that are not “essentially leak-tight” as described in 10 CFR 50 Appendix A GDC 16 for LWRs.

   Other staff and Commission discussions of containment functional performance criteria and use of containment designs other than those incorporating an “essentially leak-tight barrier” can be found in SECY-03-0047, “Policy Issues Related to Licensing Non-light-water Reactor Designs” and its SRM, SECY-05-006, “Second Status Paper on the Staff's Proposed Regulatory Structure for New Plant Licensing and Update on Policy

Issues Related to New Plant Licensing", and the NRC's Advanced Reactor Policy Statement (FR Vol. 73, No. 199, October 2008).

The DOE team considers the functional containment approach taken in ARDC 16 to be consistent with these staff and Commission positions.

b. ARDC 16 was not developed to imply anything regarding performance of an "essentially leak-tight" containment for controlling radioactive releases to the environment. ARDC 16 was written to address the positions taken by the staff and the Commission in documents referred to in response to Part a of this staff question. From the perspective of functional containment, applicants that suggest use of a "controlled release" from "essentially leak-tight containment" would need to demonstrate that the approach meets appropriate functional performance criteria.

   On the basis of information available to the DOE team during design criteria development, the need for additional design guidance that specifically addresses Part 20 compliance was not identified. As already noted in the staff question, ARDC 16, reactor functional containment "*…shall be provided to control the release of radioactivity to the environment and to assure that functional containment design conditions important to safety are not exceeded for as long as postulated accident conditions require."* This performance-oriented requirement was written to parallel the existing GDC focus on 10 CFR 50.34 and 10 CFR 52.79, which applies to design basis events rather than Part 20.

2. ARDC 25 & SFR-DC 25

The current draft version of American Nuclear Society (ANS) 54.1 (Nuclear Safety Criteria and Design Process for Sodium Fast Reactor Nuclear Power Plants) proposes the following revised language for SFR-DC 25 in the ANS 54.1 standard:

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 single malfunction of the reactivity control systems ~~such as accidental withdrawal (not ejection or dropout) of control rods.~~

Would ARDC 25 be modified in a similar way?

DOE Response:

The DOE team agrees with the strikeouts as shown in the proposed ANS 54.1 revision for ARDC 25 and SFR-DC 25.

3. ARDC 34

The ARDC combines GDC 34 and 35. However in doing so the requirement for "suitable containment capabilities" in GDC 35 was deleted. Was the deletion intentional and if so what is the basis for the deletion?

DOE Response:

The DOE team has observed that the “suitable containment capabilities" language is included GDC 35, but is not included in GDC 34. When the GDCs were being combined to form proposed ARDC 34, the “suitable containment capabilities" phrase was considered, but not retained. However, the basis for the original inclusion of this language in GDC 35 is not clear.

4. ARDC 38

Proposed ARDC 38 states*, “suitable redundancy in components and features* ***(including electric power systems operations)****, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure.”* Should the phrase in parentheses be added to incorporate descriptions of onsite and offsite power systems for SFRs being built and in operation?

DOE Response:

The ARDC description requiring redundancy includes components to assure that the safety function can be accomplished, implicitly including electric power systems. However, the phrase, "including electric power systems" can be added within the brackets to include suitable electric power systems. The word "operations" should be left out since the criterion addresses the components and features, not their operations.

5. SFR-DC 26

SFR-DC 26 states at least two independent reactivity control systems of different design principles shall be provided. According to 6.1.3 on page 22, two diverse scram systems are provided, a gravity driven rod drop and a powered rod drive-in. Figure 5, Core Layout on page 21 shows the control rod layout. Does the DOE believe that the example provided meets the intent of different design principles?

DOE Response:

Section 6.1.3 of the DOE report provides a publically available example of how a single SFR vendor has indicated their reactor design might provide reactivity control system redundancy and capability. This information is provided for general background information supporting the development of the SFR-DC provided in the DOE report. The DOE review team did not make a positive or negative judgment on any aspects of the adequacy of the design. It would be the responsibility of any SFR vendor to prove the adequacy of their reactivity control system to the NRC staff.

6. SFR-DC 28

a. Why is rod ejection (unless by positive means) deleted? For the GE-Hitachi Power Reactor Innovative Small Module (PRISM) design, features are in place to prevent rod ejection by a positive means but this may not be the case for all SFR designs.

b. Is control rod withdrawal considered an anticipated operational occurrence (AOO) or postulated accident for a SFR design? If it is a postulated accident, why is control rod withdrawal not included?

c. Does the DOE believe the SFR-DC 28 addresses the impact of a steam line break if no intermediate coolant loop exists as stated in SFR-DC 70, "***If** an intermediate coolant loop system is provided*?"

DOE Response:

a. The SFR designs evaluated by the DOE team operate at low pressure and low flow. Typically, the weight of an SFR rod bundle will exceed any uplifting force provided by the combination of core flow and pressure. Therefore, the driving function for a rod ejection is missing for the group of SFR designs that were reviewed. Rod ejection as a dominant reactivity accident is more relevant to PWRs and not relevant to SFRs. This is why rod ejection is eliminated from the list of dominant SFR reactivity accidents. However, SFR-DC 28 still requires that 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 [primary coolant] boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures, or other reactor vessel internals to impair significantly the capability to cool the core. Therefore, it would continue to be the responsibility of the vendor to prove to the NRC staff that the dominant reactivity limiting accidents have been evaluated.

b. The categorization of a control rod withdrawal as an AOO or a postulated accident will depend on the individual SFR design proposal and the associated risk assessment.

c. The proposed wording of SFR-DC-70 can be revised to address NRC Staff Question 6c. See the response to NRC Staff Question 15a.

7. SFR-DC 34

a. Proposed addition to GDC 34 includes addressing postulated accidents by adding, "provide continuous effective core cooling during postulated accidents" but does not provide details on how effective core cooling is achieved like GDC 35 does for LWRs. GDC 35 states that effective core cooling is preserved by limiting fuel and clad damage; how is effective core cooling defined for SFRs and why is it not defined in SFR-DC 34?

b. According to NUREG-1368, GDC 34, page 3-41, item 4, "any fluid in the residual heat extraction system that is separated by a single passive barrier shall not be chemically reactive with the reactor coolant." Why is this requirement not part of SFR-DC 34? Does the DOE believe that the PRISM design satisfies this GDC with respect to the reactor vessel auxiliary cooling system and auxiliary cooling system decay heat removal systems? Note that this requirement is stated for the intermediate heat transport system in SFR-DC 70.

DOE Response:

a. The definition of effective core cooling for SFRs is the same as for LWRs. SFR-DC 34 could state this explicitly by changing the last sentence of the first paragraph of SFR-

DC 34 to read, "...including anticipated operational occurrences and to prevent fuel and clad damage that could interfere with continuous effective core cooling during postulated accidents."

b. The DOE team agrees with the addition to SFR-DC 34 of the requirement of NUREG-1368, GDC 34, Item 4 that "any fluid in the residual heat extraction system that is separated from the reactor coolant by a single passive barrier shall not be chemically reactive with the reactor coolant." Note that Item 4 also references ANSI/ANS-54.1-1989 with an additional requirement to "keep the working fluid of the heat removal system at a higher pressure than the reactor coolant system." The DOE team also agrees with this requirement.

   The DOE team has not reviewed the PRISM residual heat removal system or auxiliary cooling system designs in detail. However, the DOE team believes that the PRISM design satisfies this design criterion with respect to the reactor vessel auxiliary cooling system and auxiliary cooling system decay heat removal systems.

8. SFR-DC 39

SFR passive heat removal systems require containment penetrations to accomplish their job. Should containment penetrations be included as shown (in bold) in SFR-DC 39?

"*The containment heat removal system shall be designed to permit appropriate periodic inspection of important components, such as [piping,* ***containment penetrations, etc.****] to assure the integrity and capability of the system."*

DOE Response:

The addition of "containment penetrations, etc." within the bracketed text is acceptable.

9. SFR-DC 40

Since SFR containments are similar to traditional LWR containments, the leak-tight integrity of the containment is important, as is its pressure response vs. design pressure. Should pressure response and leak-tight integrity be included as shown (in bold) to SFR-DC 40?

"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 system components, 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 associated systems."*

DOE Response:

The term "functional testing" as used in the ARDC and SFR-DC would reasonably include typical testing such as pressure tests and leak-tight integrity. Adding these terms tends to presuppose an LWR-like containment, which may unnecessarily constrain SFR designs. Therefore, the DOE team does not agree with the suggested change.

10. SFR-DC 41

For ARDC 41 should the phrase in parentheses be added to ARDC 41 to incorporate descriptions of onsite and offsite power systems for SFRs being built and in operation? *"suitable redundancy in components and features **(including electric power systems operations)**, and suitable interconnections, leak detection, isolation, and containment capabilities shall be provided to assure that the system safety function can be accomplished, assuming a single failure."*

DOE Response:

The SFR-DC description requiring redundancy includes components to assure that the safety function can be accomplished, implicitly including electric power systems. However, the phrase, "including electric power systems," can be added within the brackets to include suitable electric power systems. The word "operations" should be left out since the criterion addresses the components and features, not their operations.

11. SFR-DC 43

Since SFR containments are similar to traditional LWR containments, the leak-tight integrity of the containment is important, as is its pressure response vs. design pressure. Should pressure response and leak-tight integrity be included as shown (in bold) in SFR-DC 43?

*"The containment atmosphere cleanup 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 system components, 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 associated systems."*

DOE Response:

The term "functional testing" as used in the ARDC and SFR-DC would reasonably include typical testing such as pressure tests and leak-tight integrity. Adding these terms tends to presuppose an LWR-like containment, which may unnecessarily constrain SFR designs. Therefore, the DOE team does not agree with the suggested change.

12. SFR-DC 54

SFR-DC 54 includes the phrase, "When isolation valves are required" on pages 91 and 92 of the DOE Report. This discussion is also included in Section 6.2.5, Reactor Containment (SFR Design Criteria 50-57), of the DOE Report on page 28. Why did the DOE include the phrase "when isolation valves are required" in SFR-DC 54, rather than including a requirement for isolation valves with the understanding that an applicant may propose an exemption from this requirement where justified?

DOE Response:

All responses received to the DOE's Request for Information regarding SFR designs anticipate that containment isolation valves will not be included on all piping systems that penetrate containment. It can be deliberated that isolation valves associated with passive residual heat removal systems render the safety system less reliable. The change in wording for SFR-DC 54 “performance capabilities necessary to perform the containment safety function” allows each vendor to make a safety case to NRC staff to allow piping systems to penetrate containment without isolation valves. This avoids the need associated with an exemption request that has been a fundamental element of the DOE approach. If the safety case does not support the elimination of isolation valves, then the last sentence of SFR-DC 54 includes the specific requirements for testing operability of the isolation valve. The revised lead-in to the last sentence of the design criteria provides consistency with the text revisions proposed prior to this sentence.

13. SFR-DC 55

For a pool-type SFR design as is the case with PRISM, Super-PRISM, Toshiba (super, safe, small and simple) 4S, Experimental Breeder Reactor (EBR) II, etc., the intermediate heat exchanger (IHX) also resides within the reactor vessel (RV). The lines going from the RV to the containment would also include the intermediate heat transfer system. Why is “primary coolant” and not “primary and intermediate coolant” included in the brackets for the title and first sentence of SFR-DC 55?

DOE Response:

The title of SFR-DC 55 is the “Reactor [primary coolant] boundary.” The SFR intermediate loop is a separate closed system that does not allow any direct mixing of intermediate fluid with the primary coolant sodium. The tubing of the IHX and associated intermediate loop piping inside the RV are a part of the reactor primary coolant boundary. SFR-DC 57, “Closed system isolation valves,” addresses closed systems that penetrate containment and is the appropriate place to address a closed system, such as an intermediate loop, that penetrates containment and is not part of the reactor primary coolant boundary (in its entirety). This is similar to the treatment of the main steam system and the steam generator in a PWR.

14. SFR-DC 57

For a pool-type SFR design as is the case with PRISM, S-PRISM, 4S, EBR II, etc., the IHX also resides within the RV. The lines going from the RV to the containment would also include the intermediate heat transfer system. Why is “primary coolant” and not “primary and intermediate coolant” included in the brackets for first sentence of this SFR-DC 57?

DOE Response:

The SFR intermediate loop is a closed system. Though an SFR intermediate loop and the reactor primary coolant system share some common boundaries, they are separate systems. SFR-DC 57, “Closed system isolation valves,” specifically addresses closed systems that penetrates containment and is not part of the reactor primary coolant boundary. This is similar to the treatment of the main steam system and the steam generator in a PWR.

15. SFR-DC 70

The initial phrase of the first sentence of proposed SFR-DC 70 states “If an intermediate coolant system is provided.” This phrase would only be needed to account for an SFR design without an intermediate heat transport loop. No vendor has shared a plan to develop an SFR design without an intermediate coolant loop with the NRC.

a. What is the basis for accounting for an SFR design that does not use an intermediate heat transport loop?

b. Assuming industry were to propose an SFR design without an intermediate heat transport loop, propose an SFR-DC that requires:

    i. coolant for the secondary loop that is not chemically reactive with sodium, and

    ii. passive features to prevent overpressurization of the primary coolant boundary and pressure-pulse induced damage to SSCs within the primary coolant system in the event of a breach in the single passive barrier between primary coolant and secondary coolant.

c. Discuss the rationale for making the title plural, instead of singular.

DOE Response:

a. Although some novel SFR designs with no intermediate coolant system have been proposed in the past, all of the SFR responses received to the DOE's Request for Information rely on use of an intermediate coolant system to separate the primary coolant and the energy conversion systems. The DOE team agrees that the lead-in phrase "If an intermediate coolant system is provided" can be removed from SFR-DC 70 and replaced with the following sentence: "An intermediate coolant system shall be provided." The second sentence would begin as, "The intermediate system shall be compatible...."

b. The DOE team agrees that lead-in the phrase "If an intermediate coolant system is provided," can be removed from SFR-DC 70 and replaced with the following sentence: “An intermediate coolant system shall be provided.” (See the response to NRC Staff Question 15a). Therefore, there is no need to propose an additional SFR-DC to address this issue.

c. The DOE team agrees that making the title singular would be more appropriate.

16. SFR-DC 70

The remainder of the first sentence of proposed SFR-DC 70 states “the intermediate coolant shall be compatible with sodium if it is separated from the reactor primary coolant by a single passive barrier.” No vendor has shared a plan with the NRC to develop an SFR design that uses anything other than a single passive barrier between the primary coolant and the intermediate coolant.

a. Describe the intended meaning of the phrase “compatible with sodium.”

b. What is the basis for accounting for an SFR design that uses other than a single passive barrier between primary coolant and intermediate coolant?

c. Assuming there was a proposed SFR design that uses other than a single passive barrier between primary coolant and intermediate coolant, identify the most appropriate ARDC that would apply, or propose an SFR-DC with appropriate limiting design objectives for the barrier.

DOE Response:

a. The phrase "compatible with sodium" is intended to address chemical compatibility of the intermediate coolant with the primary sodium coolant to avoid adverse effects of such reactions on the capability of SSCs to perform their intended safety functions in case of failure of the single passive barrier that separates them.

b. An SFR design with double-wall tube intermediate heat exchanger can provide the flexibility to use an intermediate system coolant other than only those that are compatible with the primary sodium coolant. It can also be used to avoid the need for pressure differential between the primary and intermediate coolant systems.

c. The limiting design objective for the barriers between the primary and intermediate coolants is maintaining the integrity of the primary coolant boundary. SFR-DC-70 would still apply to an SFR design that uses other than a single passive barrier between primary and intermediate coolants to achieve this objective.

17. SFR-DC 70

The initial phrase of the second sentence of proposed SFR-DC 70 states, "Where a single barrier separates the reactor primary coolant from the intermediate coolant." Describe what is meant by the phrase "single barrier?" Regarding items 16 b and 16 c (above), are there SFR designs under consideration that have double walled primary coolant to intermediate coolant heat exchanger tubes? Would they be characterized as "a double passive barrier?"

DOE Response:

Single barrier is meant as a single-wall tube (as opposed to a double-wall tube) in an intermediate heat exchanger. Although past and current SFR designs rely on use of a single passive barrier to separate the primary and intermediate coolant systems, an SFR design with a double-wall tube intermediate heat exchanger cannot be ruled out (see the response to NRC Staff Question 16b). The DOE team believes that a double walled/tubed intermediate heat exchanger would be characterized as a double passive barrier.

18. SFR-DC 70

The remainder of the second sentence of proposed SFR-DC 70 states, "a pressure differential shall be maintained such that any leakage would flow from the intermediate coolant system to the reactor primary coolant system unless other provisions can be shown to be acceptable." Please describe what other provisions are envisioned and the primary rationale for their acceptability.

Staff notes that this phrase is apparently inherited from NUREG-0968, pages 3-19 & 3-20, which presents Clinch River Breeder Reactor (CRBR) design Criterion 31—Intermediate Coolant System, as follows (CRBR Criterion 31 with apparently inherited language highlighted in bold):

The intermediate coolant system shall be designed to transport heat reliably from the reactor coolant system to the steam/feedwater systems as required for the reactor coolant system to meet its safety functions under all plant conditions of normal operation, including anticipated operational occurrences, and postulated accident conditions. ***The intermediate coolant*** *system shall contain coolant that* ***is not chemically reactive*** *with the reactor coolant.*

A pressure differential shall be maintained across *a passive boundary between the reactor coolant system and the intermediate coolant system so* ***that any leakage would*** *tend to* ***flow from the intermediate coolant system to the reactor coolant system unless it can be shown that other provisions are acceptable*** *on some defined basis.*

DOE Response:

The provisions other than a pressure differential between the intermediate and primary coolant systems to prevent the leakage of the primary coolant into the intermediate coolant system will be dependent on the design. However, a double-wall tube intermediate heat exchanger can be considered as a potential provision to eliminate the need for pressure differential.

19. SFR-DC 70

The third sentence of proposed SFR-DC 70 states, "*The intermediate coolant boundary shall be designed to permit inspection and surveillance in areas where leakage can affect the safety functions of systems, structures and components.*"

a. Please describe what testing is envisioned by the term "surveillance" with respect to the intermediate coolant boundary. Is the intended meaning a material surveillance program?

Staff notes that this sentence is apparently partially inherited from CRBR design Criterion 33—Inspection and Surveillance of Intermediate Coolant Boundary, as follows (CRBR Criterion 33 with apparently inherited language highlighted in bold):

Components that are part ***of the intermediate coolant boundary shall be designed to permit*** *(1) periodic* ***inspection*** *of* ***areas*** *and features important to safety 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 intermediate coolant leakage.*

If the intended meaning is a "material surveillance program" as indicated above by item (2), why was "material" excluded from SFR-DC 70?

b. Since the intermediate coolant system interfaces not only with the primary coolant system, but also the tertiary coolant system, was consideration given to including provisions regarding the (1) intermediate coolant system interface with the tertiary coolant system; and (2) selection of intermediate boundary areas for "inspection and surveillance" based on potential impacts of tertiary system coolant leakage into the intermediate coolant system?

DOE Response:

a. Leak detection, intermediate coolant activity monitoring, intermediate coolant system inventory, and pressure monitoring can be considered as part of a surveillance program. The DOE team considers the term "material surveillance program" too specific and more narrowly defined than the term "surveillance program".

b. No SFR designs suggest the use of a tertiary coolant system. Intermediate coolant system interfaces with the energy conversion system. Reaction between the intermediate coolant and working fluid of energy conversion system is addressed in SFR-DC 74 (within the context of sodium-water reactions).

20. SFR-DC 70

Based on assuming that SFR-DC need only address SFR designs with an intermediate cooling system where primary and intermediate coolants are separated by a single passive barrier, and considering 19 a and b (above), are the additions and deletions to SFR-DC 70 as depicted below appropriate?

> *Intermediate coolant systems*
> *~~If an~~ **An** intermediate coolant system ~~is provided, the intermediate coolant shall be compatible with sodium if it is and~~ **shall be provided.** ~~separated from the~~ **A single passive barrier shall separate intermediate coolant from** reactor primary coolant; **at least a single passive barrier shall separate tertiary coolant from intermediate coolant** ~~by a single passive barrier~~. **The intermediate coolant shall be chemically nonreactive with sodium.** ~~Where a single barrier separates the reactor primary coolant from the intermediate coolant, a~~ **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 reactor primary coolant system ~~unless other provisions can be shown to be acceptable~~. The intermediate coolant boundary shall be designed to permit ~~inspection and~~ **the conduct of a material** surveillance **program and inspection** in areas where **intermediate coolant** leakage **out of the intermediate coolant system, or tertiary coolant leakage into the intermediate coolant system,** ~~can~~ **may hinder or prevent** ~~affect the safety functions of systems,~~ **a** structure~~s~~**, system,** ~~and~~ **or** component~~s~~ **from performing any of its intended safety functions**.*

With additions and deletions applied, SFR-DC 70 would state:

> *Intermediate coolant system*
> *An intermediate cooling system shall be provided. A single passive barrier shall separate intermediate coolant from reactor primary coolant; at least a single passive barrier shall separate tertiary 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 reactor primary coolant system. The intermediate coolant boundary shall be designed to permit the conduct of a material surveillance program and inspection in areas where intermediate coolant leakage out of the intermediate coolant system, or tertiary coolant leakage into the intermediate coolant system, may hinder or prevent a structure, system, or component from performing any of its intended safety functions.

DOE Response:

Although the text proposed by NRC generally captures the intent of SFR-DC 70, it appears to stipulate the use of a single passive barrier between the primary and intermediate coolant systems, and extend the requirement for intermediate coolant system boundary integrity to include the steam generator. It also narrowly defines the surveillance program as "material surveillance program." The DOE team prefers the proposed SFR-DC 70 language (see the response to NRC Staff Question 15a), which also reflects the stakeholder feedback.

21. SFR-DC 71

SFR-DC 71 appears to be based on NUREG-0968 (page 3-21), CRBR design Criterion 34— Reactor and Intermediate Coolant and Cover Gas Purity Control, with differences as shown by the following CRBR and SFR-DC 71 design criterion:

CRBR Criterion 34 - *Systems shall be provided to monitor and maintain reactor, intermediate coolant, and cover gas purity within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, (3) radionuclide concentrations, and (4) detection of sodium-water reactions.*

SFR-DC 71 - *Systems shall be provided as necessary to maintain primary coolant purity and cover gas purity within specified design limits. These limits shall be based on consideration of (1) chemical attack, (2) fouling and plugging of passages, and (3) radioisotope concentrations.*

a. Please discuss why intermediate coolant purity is not addressed by an SFR-DC.

b. Please discuss why the term “sodium” is not explicitly used; e.g., “purity of primary coolant system sodium and cover gas,” since SFRs use sodium.

c. Please discuss why the term “radioisotope” was used in place of “radionuclide” which is the term used for all GDC, mHTGR-DC, and all other SFR-DC in the DOE report.

DOE Response:

a. The DOE team, also based on the stakeholder feedback, considers the intermediate coolant purity control an operational issue.

b. There is no specific reason for not explicitly using the term "sodium." It can be included.

c. There is no specific reason for using the term "radioisotope." It can be replaced with "radionuclide".

22. SFR-DC 72

The rationale for SFR-DC 72 provided on page 97 of the DOE report states:

> *NUREG-1368 (page 3-56) (ML063410561) Section 3.2.4.2 suggested the need for a separate criterion for sodium heating systems. Also, a separate criterion was included in NUREG-0968 (ML082381008) (CRBR design Criterion–7 Sodium Heating Systems).*

The differences between SFR-DC 72 and NUREG-0968 (page 3-10) CRBR design Criterion 7—are shown in the following:

CRBR Criterion 7 - *Heating systems shall be provided as necessary for systems and components important to safety that contain, or may be required to contain, sodium or sodium aerosol. The heating systems and their controls shall be appropriately designed with suitable redundancy to ensure that the temperature distribution and rate of change of temperature in sodium systems and components are maintained within design limits assuming a single failure. The heating system shall be designed so that its failure will not impair the safety function of associated systems and components.*

SFR-DC 72 - *Heating systems shall be provided as necessary 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.*

The justification in NUREG-0968 states in part that the "intent of this criterion is to require that systems important to safety that contain sodium or sodium aerosols and that require a controlled temperature for the system to perform its safety function be provided with a heating system capable of ensuring that desired temperatures are maintained and designed to preclude overheating the components to which they are attached…external heat is required to be supplied to the sodium systems under certain plant conditions to keep the sodium molten and to keep sodium aerosol from condensing and plugging flow paths exposed to sodium vapor."

NUREG-1368 Section 3.2.4.2 Sodium Heating Systems (page 3-56) refers to CRBR design Criterion 7, and states "The intent of the criterion is to require that systems important to safety, and which contain sodium or sodium aerosols and require a controlled temperature for the system to perform its safety function, be designed and maintained to preclude overheating (creating aerosols) and underheating (condensing aerosols and freezing sodium) the system." It also states "Requirements for system features similar to those listed in SRP Section 9.3.4, Item III.A.9 (Ref. 3.9), should be developed for sodium systems in LMRs."

a. SFR-DC 72 does not appear to require designing the heating system to prevent it from overheating sodium. Discuss why this is not a design concern that should be more explicitly accounted for in an SFR-DC.

b. SFR-DC 72 does not appear to address providing external heat to the primary coolant sodium cover gas system to maintain cover gas temperature within limits. Discuss why keeping sodium aerosol (or vapor) from condensing and plugging flow paths exposed to sodium vapor is not a design concern that should be explicitly accounted for in an SFR-DC.

c. Discuss why proposed SFR-DC 72 does not use the terms “sodium aerosol” and “sodium vapor.”

d. Discuss why proposed SFR-DC 72 contains no language corresponding to the last sentence of NUREG-0968 CRBR design Criterion 7, which states “The heating system shall be designed so that its failure will not impair the safety function of associated systems and components.”

e. Discuss whether portions of the sodium heating system would be required following a postulated event for accident mitigation and fuel protection. If sodium heating would be needed following a postulated event, which proposed ARDC and SFR-DC would ensure that the SFR design will provide sufficient onsite electrical power to support operation of credited sodium heating systems?

DOE Response:

a. SFR-DC 72 stipulates use of heating systems and their controls to assure that the temperature distribution and rate of change of temperature in systems and components important to safety are maintained within design limits. The DOE team considers the phrase "within design limits" to adequately address both sodium freezing and overheating conditions.

b. In certain SFR designs, maintaining temperature controls on lines containing cover gas and the sodium aerosol/vapor contained in the cover gas can be important to safety. Therefore, SFR-DC 72 should be modified to state that if plugging of any cover gas line due to condensation or plate out of sodium aerosol or vapor could prevent a safety function from being accomplished, the temperature control associated with that line shall be considered "important to safety.”

c. See response to NRC Staff Question 22b.

d. SFR-DC 72 requires the sodium heating systems to assure that the systems and components containing sodium maintain their safety function assuming a single failure. The DOE team considers that the "single failure" clause addresses the CRBR requirement for the heating system failure to not impair the safety function of associated systems and components.

e. Most (if not all) postulated accidents lead to elevated temperatures, not requiring the sodium heating system function. However, the DOE team cannot not rule out a need for sodium heating system following a postulated event since the answer would be design-specific and would depend on selection of the licensing basis events. Nevertheless, during a postulated event that might require sodium heating system function, the power system requirement is addressed in SFR-DC 17.

23. SFR-DC 73

The rationale for SFR-DC 73 and SFR-DC 74 provided on page 97 of the DOE Report states:

> *NUREG-1368 (page 3-56) (ML063410561) Section 3.2.4.1 [Protection Against Sodium Reactions] 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).*

Content in NUREG-0968 (page 3-10) CRBR design Criterion 4 and SFR-DC 74 are shown below:

> CRBR Criterion 4 - *Systems, components, and structures containing sodium or NaK shall be designed and located to limit the consequences of chemical reactions resulting from a sodium or NaK spill. Special features such as inert atmosphere vaults shall be provided as appropriate for the reactor coolant system. Fire-control systems and means to detect sodium, NaK, or their reaction products shall be provided to limit and control the extent of such reactions to ensure that the functions of components important to safety are maintained. Means shall be provided to limit the release of reaction products to the environment, as necessary, to protect plant personnel and to avoid undue risk to the public health and safety. Material that might come in contact with sodium or NaK shall be chosen to minimize the adverse effects of possible chemical reactions or microstructural changes. In areas where sodium or NaK chemical reactions are possible, structures, components, and systems important to safety, including electrical wiring and components, shall be designed and located so that the potential for damage by sodium chemical reactions is minimized. Means shall be provided as appropriate to minimize possible contacts between sodium/NaK and water. A single failure of a passive boundary shall not permit the contact of primary coolant with water/steam. The effects of possible interactions between sodium/NaK and concrete shall be considered in the design.*
>
> SFR-DC 73 - *Means to detect sodium leakage and to limit and control the extent of sodium-air and sodium-concrete reactions shall be provided as necessary 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 as appropriate for systems containing reactor primary sodium coolant.*

a. Discuss why SFR-DC 73 includes no language specifically addressing means to detect, control, and extinguish a fire resulting from leaked sodium reacting with air, concrete, or water.

b. Discuss why SFR-DC 73 includes no language specifically requiring (as appropriate) means to detect sodium leakage or special features such as inerted enclosures or guard vessels for the intermediate coolant system or any system containing sodium.

DOE Response:

a. The DOE team intended that the phrase “detect sodium leakage and to limit and control the extent of sodium-air and sodium-concrete reactions” includes extinguishing the resulting fire. However, the explicit inclusion of the phrase “and to extinguish a fire as a result of these sodium-air and sodium-concrete reactions” can be added for clarity. The sodium-water reaction is included in SFR-DC 74.

b. SFR-DC 73 is not restricted to primary coolant and can include intermediate or residual heat removal system coolants. In general, sodium fires are considered operational issues. SFR-DC 73 is intended to assure that the safety functions of structures, systems and components important to safety are maintained in case of sodium fires.

24. SFR-DC 74

SFR-DC 74 contains criteria for sodium/water reaction prevention and mitigation. Considering a and b (below), are the additions and deletions to SFR-DC 74 as depicted below appropriate?

a. The source of molten sodium that could potentially come into contact with water or steam may not be limited to systems considered “important to safety.” Discuss why this criterion is limited to sodium-containing SSCs that are considered “important to safety.”

b. The term “consequences” has a connotation of estimated radiological dose from radionuclides released during an event, such as a design basis accident. Discuss why this SFR-DC does not focus on the more immediate adverse impact of a sodium and water reaction on operability of SSCs with required safety functions.

Structures, systems, and components ~~important to safety~~ containing sodium shall be designed and located to limit the ~~consequences~~ ***adverse effects of chemical reactions between sodium and water*** *on* ***the capability*** *of ~~safety functions of~~ any ~~systems, s~~tructures,* ***system, or*** *~~and~~ component~~s~~* ***to perform any of its intended safety functions****. Means shall be provided as appropriate to limit possible contacts between sodium and water.*

DOE Response:

a. In general, sodium fires and sodium-water reactions are considered operational issues unless they impact the safety function of SSCs. However, the DOE team agrees with NRC’s suggestion that the first sentence in proposed SFR-DC 74 can be revised as follows without expanding its scope:

"Structures, systems, and components ~~important to safety~~ containing sodium shall be designed and located to limit the ~~consequences~~ adverse effects of chemical reactions between sodium and water on the ~~safety functions~~ capability of any ~~systems,~~ structure~~s~~, system, or ~~and~~ component~~s~~ to perform any of its intended safety functions."

b. In general, adverse effects of sodium fire and sodium-water reaction are considered operational issues unless they impact the safety functions of SSCs and lead to radiological consequences. However, the DOE team agrees with NRC suggestion that the first sentence in proposed SFR-DC 74 can be revised as shown above.

25. SFR-DC 74

SFR-DC 74 contains criteria for sodium/water reaction prevention and mitigation. The second paragraph specifically discusses the sodium-steam generator system.

a. Was the possibility of a coolant medium other than water (i.e. helium, or other inert gas) in the tertiary system considered along with the use of the Brayton Cycle for power conversion? Would a new SFR-DC addressing a non-water tertiary coolant system be appropriate?

b. Would a new SFR-DC addressing the higher pressures of the Brayton Cycle power conversion system be appropriate?

c. In consideration of the phrase from the second paragraph, "the sodium-steam generator system shall be designed to . . . limit the effects of the energy and reaction products released," when would "the effects" be beyond the "limit"? Consider that the limit could be in the range from (i) one redundant subsystem degraded but operable, to (ii) loss of a redundant subsystem with the other redundant subsystem degraded but operable (recognizing that a passive safety system may not be redundant).

d. Why does the initial phrase of the second paragraph of SFR-DC 74, "If necessary to prevent loss of any plant safety function," need to be included? If the tertiary system uses high pressure steam and water and the intermediate system coolant is sodium (or other medium that reacts strongly with water), then the heat transfer interface between the intermediate and tertiary heat transport systems must be designed consistent with the stipulations of the remainder of the paragraph (except, see item c above).

DOE Response:

a. All of the SFR responses received to the DOE's solicitation of input from SFR designers are based on steam-cycle for the energy conversion system. Although there is an interest in alternative energy conversion systems based on Brayton cycle using super-critical CO_2 or Ni gas as the working fluid, these systems remain under development.

   The SFR-DC set is intended to demonstrate how proposed adaptations of GDC for advanced reactors can be utilized to guide the development of a design-specific PDC. If a designer chooses to utilize an alternative energy conversion system instead of the conventional steam-cycle, a new criterion addressing the potential impact of chemical reactions between the sodium and alternative working fluid can be considered as part of their PDC.

b. Please see the response to NRC Staff Question 25a. If the designers choose to utilize an alternative energy conversion system instead of the conventional steam-cycle, a new criterion addressing the impact of potentially higher pressures of the energy conversion system can be considered as part of their PDC.

c. In the case of the sodium-steam generator system, the "limit" implies protecting the integrity of the primary coolant system by avoiding a breach of intermediate heat exchanger tubes.

d. Capabilities to detect and contain sodium-water reactions in the sodium-steam generator system are considered necessary to protect the integrity of the primary coolant system from the effects of the energy and reaction products released by such reactions. Therefore, the words "If necessary" can be removed from the beginning of the second paragraph of proposed SFR-DC 74.

26. HTGR-DC 5

The DOE report specifically modified GDC 5 for mHTGR-DC 5 to replace “nuclear power unit” with “reactor module” and “nuclear reactor units” with “module groups.” Further, mHTGR-DC 5 includes the sentence *“SSCs important to safety shall not be shared among reactor modules or reactor module groups.”* Was it the DOE’s intent to imply that sharing within a reactor module group is acceptable? Also there are other advanced non-LWR designs that feature modularity (e.g., PRISM). Why was modularity considered only for the mHTGR design?

DOE Response:

It was not DOE’s intent to imply that sharing SSCs important to safety within a reactor module group is acceptable. The use of the words “shall not be shared among reactor modules” was primarily intended to address sharing within a reactor module group. The possibility of sharing between reactor modules belonging to two different reactor module groups was not considered in the development of modular HTGR-DC 5 because there would be no design incentive to share SSCs important to safety in that manner. However, such sharing would also be precluded by the words chosen for the proposed criterion.

Regarding other advanced non-LWR designs that feature modularity, the modular HTGR is an advanced non-LWR that has been consistently designed to incorporate modularity. Therefore, issues associated with modularity were directly addressed in the modular HTGR design criteria. In the case of the SFR, modularity is provided for some designs, but it is not provided for others. It was decided to prepare the SFR design criteria for a traditional single-unit configuration. If other advanced non-LWR designs incorporating modularity are submitted to NRC, it would be expected that modularity issues would be addressed in the proposed principal design criteria as appropriate. However, it should be noted that modular HTGR-DC 5 is intended to address these issues only for the modular HTGR. Other advanced non-LWR modular designs may choose to use different terminology and/or wording in developing their design criteria.

27. HTGR-DC 10

The intent of GDC 10 is to set limits and/or design protection systems with appropriate margin to protect against fuel failures caused by events likely to occur during normal operation in the lifetime of the plant. The specified acceptable core radionuclide release design limit (SARRDL) approach appears to accommodate potential AOO caused fuel failures and does not set a specified acceptable fuel design limit (SAFDL) to preclude them. How would the proposed SARRDL approach preclude additional fuel failures caused by an AOO?

DOE Response:

With the exception of a few design-specific changes in terminology, the overall wording and structure of the proposed modular HTGR-DC 10 is the same as that of GDC 10 in 10 CFR 50 Appendix A. As is the case for the SAFDL provided in GDC 10 of 10 CFR 50 Appendix A, the SARRDL is not to be exceeded during normal operation or AOOs. No difference is intended in the most important underlying objective of the modular HTGR criterion relative to that of GDC 10 in Appendix A, ensuring that the dose to the public remains within regulatory limits.

In light water reactors, this objective is met by ensuring that there is no incremental fuel failure (i.e., fuel rod cladding failure) during AOOs. However, there are billions of coated fuel particles in a modular HTGR core. Accordingly, it is not statistically possible (and, as discussed below, not necessary) to ensure that there will be zero coated fuel particle failures during normal operation and AOOs.

Because the radionuclide inventory in each coated fuel particle is very low, the safety significance of a single coated fuel particle failure is much less than that of a single LWR fuel rod failure. Furthermore, radionuclide release from coated fuel particles is a complex function of the condition of each of the particle coating layers and the radionuclide species under consideration. Radionuclide release cannot be correlated to a small number of simple fuel performance parameters such as peak fuel temperature, maximum neutron fluence, or reactor power to flow ratio. Use of the SARRDL to address GDC 10 by providing a specified acceptable core radionuclide release design limit for the modular HTGR is appropriate because the SARRDL is directly related to the overall fuel performance in the reactor core and is directly related to potential offsite dose during and following postulated accidents. As long as the SARRDL is not exceeded during normal operation and AOOs, the objective of ensuring that dose to the public remains within regulatory limits will be met for normal operation, AOOs, and postulated accidents.

28. HTGR-DC 10

HTGR-DC 10 replaces the concept of SAFDLs with SARRDLs. HTGR-DCs 12, 17, 20, 25, 26, 34 also propose the same change. Pages 41-42 of the DOE Report describe the basis for this change and state that the SARRDL value (to be determined on a design-specific basis) will be set so that the calculated offsite doses do not exceed the regulatory requirements at the Exclusion Area Boundary for each of the most limiting licensing basis events.

a. Are only accidents considered for the SARRDLs?

b. Which regulatory requirements is DOE assuming should be met at the EAB – the 10 CFR 50.34/10 CFR 52.47 design and siting criterion (25 rem total effective dose equivalent (TEDE) over a maximum 2-hr release period), 10 CFR Part 20 limits to the public, or other regulatory requirements?

c. If only accidents and the siting and design dose criteria are used to set an acceptable SARRDL, how would this assure that the radioactivity in the system is controlled so that 10 CFR Part 20 limits to workers and the public are not exceeded for normal operation? For AOOs?

DOE Response:

GDC 10 of 10 CFR 50 Appendix A requires that the SAFDL not be exceeded during normal operation and AOOs. The same requirement is applied to the SARRDL in the proposed modular HTGR-DC 10. As noted by the staff in this question, the SARRDL is set to ensure that calculated offsite doses do not exceed the regulatory requirements at the Exclusion Area Boundary for each of the most limiting licensing basis events. The values are established, for licensing basis events involving a breach of the helium pressure boundary, taking into account the contributions to offsite dose from release of the gaseous radionuclide inventory in the helium and a fraction of the condensable radionuclide inventory within the reactor helium pressure boundary. In addition, for events also involving core heatup, a delayed release of a fraction of the radionuclide inventory in the reactor core is taken into account.

The top-level regulatory requirements to be met are as follows:

- 10 CFR 20 annualized offsite dose guidelines
  - 100 mrem/yr total effective dose equivalent
  - Measured on a cumulative basis annually at the EAB of the site
  - For normal operation and anticipated operational occurrences
- 10 CFR 50.34 (10 CFR 52.79) accident offsite doses
  - 25 rem total effective dose equivalent
  - Evaluated at the site EAB at 2 hours and at the site LPZ at 30 days
  - Design basis for off-normal events
- NRC Safety Goals individual fatality risks
  - Prompt Quantitative Health Objective (QHO) of 5×10-7/yr, latent QHO of 2×10-6/yr Evaluated at 1 mile for prompt and 10 miles for latent
  - Overall assurance of negligible cumulative risks during normal operation and off-normal events

Relative to 10 CFR 20 limits on exposure to workers, the modular HTGR SARRDL plays the same role as the large LWR SAFDL in ensuring that these limits are met. In addition to the role of the SARRDL in limiting the magnitude of the radiation source, worker exposure is controlled, as it is in all nuclear power plants, by a combination of traditional measures including shielding, standoff distance, and limits on duration of exposure.

To summarize relative to the three parts of the staff question:

a. Normal operations, AOOs, and accidents are all considered in setting the SARRDL.

b. A spectrum of top-level regulatory requirements is to be met.

c. The regulatory requirements to be met include the requirements of 10 CFR 20 for the public. The role of the SARRDL in meeting worker dose limits is analogous to that of the LWR SAFDL.

29. HTGR-DC 13

For HTGR-DC 13, the ARDC 13 was modified to remove “reactor core, [reactor coolant pressure boundary, the containment and its associated systems].” Why should the reactor core and helium pressure boundary be excluded from monitoring?

DOE Response:

The modular HTGR reactor core and reactor helium pressure boundary are part of the functional containment referred to in modular HTGR-DC 13; see Section 7.1.4 of the DOE report (INL/EXT-14-31179, Rev. 1) for a discussion of the functional containment concept. As a consequence, the reactor core and reactor helium pressure boundary are subject to monitoring due to the inclusion of functional containment within modular HTGR-DC 13.

30. HTGR-DC 20

The intent of GDC 20 is to establish protection system setpoints to prevent SAFDLs from being violated during an AOO. To prevent the SARRDL from being violated, a means of predicting additional fuel failures and associated radionuclide inventory would need to be developed for postulated AOOs and protective systems setpoints established. Does the DOE envision development of protective setpoints to ensure SARRDLs are not violated for AOOs?

DOE Response:

It is anticipated that appropriate protection system setpoints may be established to ensure that various operating limits are not exceeded. Details will be provided by future applicants.

31. HTGR-DC 25

The current draft version of ANS 54.1 (Nuclear Safety Criteria and Design Process for Sodium Fast Reactor Nuclear Power Plants) proposes the following revised language for SFR DC 25:

SFR-GDC 25 Protection system requirements for reactivity control malfunctions.

The protection system shall be designed to assure that specified acceptable fuel design limits are not exceeded during any anticipated operational occurrence for any resulting from a single malfunction of the reactivity control systems ~~such as accidental withdrawal (not ejection or dropout) of control rods.~~

Would mHTGR-DC 25 be modified in a similar way?

DOE Response:

The DOE team agrees with the strikeouts as shown in the proposed ANS 54.1 revision for modular HTGR-DC 25. However, the reference to SAFDL would still need to be replaced with SARRDL for the modular HTGR.

32. HTGR-DC 30

In the DOE Report Table 9.3, mHTGR-DC 14, “Reactor Helium Pressure Boundary” specifies that the reactor helium pressure boundary be designed, fabricated, erected, and tested with a low probability of abnormal leakage including unacceptable ingress of air, secondary coolant or other fluids. However, mHTGR-DC 30, “Quality of the Reactor Helium Pressure Boundary,” only specifies a means to detect helium leakage out of the boundary but does not address detection of air, secondary coolant or other fluids into the boundary. Why was the detection of substances leaking in not included in mHTGR-DC 30?

DOE Response:

DOE suggests that the following phrase be added at the end of modular HTGR-DC 30: *"...or unacceptable ingress of air, secondary coolant, or other fluids."*

33. HTGR-DC 33

GDC 33 requires maintaining reactor inventory such that the SAFDLs are not violated for small breaks or leaks. This is based on normal operations including AOOs. On page 111 of the report the rationale for this mHTGR-DC states the helium makeup system does not assure the SARRDLs are met by this system. Are the SARRDLs met for small leaks and breaks of the coolant boundary (i.e., a partial loss of coolant)? How would normal operation leaks and breaks meet the SARRDLs and not just the adequate cooling criterion (which is taken to mean a postulated accident criterion)?

DOE Response:

The SARRDLs are met for small leaks and breaks of the coolant boundary. Depending on the size of the leak or small break, these events may be classified as normal operations or as anticipated operational occurrences. The break size at which the event becomes a design basis event is specific to individual designs.

In GDC 33 of 10 CFR 50 Appendix A, the intent is to ensure that adequate reactor coolant inventory is maintained to provide heat removal for small leaks or breaks, thereby ensuring that SAFDLs are not exceeded. This is important for LWRs because they are relatively sensitive to undercooling scenarios.

In contrast, modular HTGRs are relatively insensitive to undercooling scenarios. In the event of a leak or small break of the reactor helium pressure boundary, the modular HTGR is designed to provide adequate heat removal independent of helium pressure, even including cases in which helium pressure is reduced to ambient conditions. More information regarding this capability can be found in INL/EXT-11-22708, "Modular HTGR Safety Basis and Approach," ADAMS Accession No. ML11251A169.

Although modular HTGRs have helium makeup and cleanup systems, they are not relied upon to ensure that adequate heat removal is maintained and that the SARRDLs are met for small leaks and breaks. Accordingly, GDC 33 is not applicable to the modular HTGR.

34. HTGR-DC 34

Why wouldn't a design requirement of the reactor cavity cooling system (RCCS) to maintain system geometry, including pressure vessel geometry, be included?

It is understood that there are other means to provide cooling during normal shutdown, but the write-up in Section 7.2.4 states the RCCS "is applicable to both normal and accident conditions," so why was "all shutdown conditions following normal operation…" deleted?

DOE Response:

The modular HTGR pressure vessel geometry and overall geometry related to passive residual heat removal are maintained by the reactor building and is addressed by modular

HTGR-DC 71. Additionally, modular HTGR-DC 70 addresses the structural design basis for the reactor vessel and reactor system relative to the overall geometry for passive heat removal. The RCCS does not play a role in preserving passive residual heat removal structural geometry.

As stated in Section 7.2.4 of the DOE report, modular HTGR-DC 34 addresses RCCS operation during both normal operations and accident conditions. This includes all conditions associated with plant shutdown following normal operations.

The phrase *"…all shutdown conditions following normal operations…"* was added to ARDC 34 in consideration of specific language contained in GDC 34, Section 3.2.3 of NUREG-1368, regarding the removal of residual heat in relation to the proposed PRISM LMR design. This document was specific to an SFR design.

For purposes of modular HTGR residual heat removal design, all shutdown conditions following normal operations are understood to be part of normal operations and are adequately addressed by the text contained in modular HTGR-DC 34. The ARDC 34 language related to *"…all shutdown conditions following normal operations…"* has been deleted on that basis.

35. HTGR-DC 54

The DOE Report, page 119 states that "the modular HTGR Reactor Building does not provide a pressure retention function, and is not relied upon to meet the offsite dose requirements of 10 CFR 50.34 (10 CFR 52.79)." However, the INL report on page 104 indicates that ARDC 16, Containment Design, applies to the mHTGR design. Discuss why ARDC 54 for piping systems penetrating containment should not continue to apply to the mHTGR.

DOE Response:

The DOE report (page 51) addresses existing GDC 16 by suggesting proposed ARDC language regarding containment design. The proposed ARDC 16 content describes the concept of a functional containment and summarizes the two basic approaches to implementation of the concept, which may "…consist of a structure surrounding the reactor and its cooling systems…" or "…multiple barriers internal and/or external to the reactor and its cooling system…". ARDC 16 is implemented without any further clarification for modular HTGRs by applying the "multiple barriers" approach for controlling the release of radionuclides, rather than the "structure surrounding the reactor" approach.

As described in the Rationale for Modification for ARDC 54, found on page 68 of the DOE report, ARDCs 51–57 apply to advanced non-LWR designs "that utilize a fixed containment structure" to meet the functional containment requirements specified in ARDC 16. Since modular HTGRs do not utilize the containment structure option to address ARDC 16, the related concept of "piping systems penetrating containment" from Question 35 above does not physically exist, and ARDCs 51–57 do not apply.

Further information regarding the multiple barrier approach implemented in modular HTGRs is provided in DOE report Sections 7.1.4 (Functional Containment) and 7.2.5 (Reactor Containment), and in the Rationale for Modification for modular HTGR-DC 50.

36. HTGR-DC 70

In the rationale for modification for mHTGR-DC 70, "reactor system" is synonymous with "reactor internals." The current design requirements for reactor internals in light water reactors are typically GDC 2, 4 and 10. No similar DC is proposed for the SFR-DCs and the ARDCs. Would mHTGR-DC 70 would still be needed if HTGR-DC 10 were modified as shown below:

The reactor ~~core~~ system and associated ***[~~coolant~~ heat removal]****,* ***structures****, 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.*

DOE Response:

The reactor system includes the core (fuel and reflector), the control rods, and those components that are analogous to reactor internals for other reactor types (structural supports, core barrel, etc.). The rationale for modular HTGR DC 70 was not intended to imply that "reactor system" is synonymous with "reactor internals," but instead to highlight the reactor internals portion of the reactor system because they are the components that are in the passive residual heat removal pathway. DOE would suggest that the rationale for modification for mHTGR-DC 70 would be clearer if it were updated as follows:

New modular HTGR design-specific GDC is necessary to assure reactor vessel and reactor system (~~reactor internals~~including the fuel, reflector, control rods, core barrel, and structural supports) integrity is preserved for passive heat removal and for insertion of neutron absorbers.

Existing GDC 10, proposed ARDC 10, proposed SFR-DC 10, and proposed modular HTGR-DC 10 all apply during any condition of normal operation, including the effects of anticipated operational occurrences. The DOE report proposed the addition of modular HTGR-DC 70 due to the importance of preserving the SSC configurations (reactor vessel and reactor system) necessary to support passive residual heat removal and neutron absorber insertion during postulated accidents.

On this basis, we have evaluated the suggested approach regarding modular HTGR-DC 10 and modular HTGR-DC 70. We have concluded that modular HTGR-DC 10 should not be modified as suggested, since the criterion would still apply only during normal operation conditions, including anticipated operational occurrences. The suggested modification would not address the underlying basis of modular HTGR-DC 70, which is to establish SSC structural design basis requirements for passive residual heat removal during postulated accidents. There is no criterion similar to modular HTGR DC 70 proposed for the SFR or for the ARDCs because the passive residual heat removal characteristics of the modular HTGR are unique to that design.

37. HTGR-DC 73

As discussed in the NRC staff's pre-licensing assessment reports for Next Generation Nuclear Plant (ML14174A626), the major accidents to be considered for determining the siting source terms for modular HTGRs may include severe air ingress accidents with significant graphite oxidation in the core and support structures. That discussion refers to the Staff Requirements Memorandum (SRM) to SECY-93-092, which specifically directs the staff to consider “chimney effect” air ingress events. The published technical literature shows that some modular HTGR developers are pursuing design features that would limit the progression of air ingress and graphite oxidation during such major accidents and also support or enable the timely termination of graphite oxidation by air.

Should design criteria for such design features be specified by either supplementing any of the currently proposed HTGR-DC or by adding a new design criterion for that purpose (e.g., HTGR-DC 73)? Are there emergency procedures or other programmatic measures that would support the effectiveness of such design features for terminating oxidation in the context of mitigating strategies?

DOE Response:

The potential for a spectrum of air ingress events in the modular HTGR is included when establishing and evaluating the event sequences considered during licensing basis event and design-basis accident selection. Past work in this area indicates that the event sequence referred in this staff question is extremely unlikely and would fall below the frequency threshold of beyond design-basis events (BDBE). In 1995, the very low probability of this sequence was acknowledged by the NRC staff and documented in Section 3.4.3.6 of NUREG-1338. Consequently, and as was noted in the question, severe air ingress event sequences will be considered as applicable when establishing the physically plausible bounding event sequences to be considered when deriving siting source terms for the modular HTGR. Modification to proposed modular HTGR design criteria is not necessary since an event sequence of this type is outside the scope of the criteria. A discussion of emergency procedures or other programmatic measures related to postulated events of this type is also considered beyond the scope of the design criteria.

38. General Question

Given that the DOE Report limits its scope to design basis accidents (DBAs), what relevant information, if any, was utilized from Reference 6 of Chapter 8 (C. Boardman, et. al., “Containment Performance of S-PRISM under Severe Beyond Design Basis Conditions” 9th International Conference on Nuclear Engineering, Paris, France, April 2001) which deals with Severe Beyond Design Basis Accidents?

DOE Response:

The PRISM reactor containment design has evolved over time. Depictions of the original PRISM containment design are no longer considered representative for the overall PRISM concept. Reference 6 included a depiction of a more recent SFR containment design for the S-PRISM reactor and basic values for design leak rate and design pressure. The figure and the basic containment design values for the S-PRISM concept were included in Section 6.1.6 of the DOE report. No consideration regarding beyond design-basis accidents was included, consistent with the scope of the existing GDCs.

39. General Question

Page xiv of the DOE Report defines the acronym “SFR” as “sodium fast reactor.” Subsequent titles in Chapter 6 of the report also refer to “sodium fast reactors.” However, on page 27, Section 6.2.2, lines 6-7, the authors state “…sodium-cooled fast reactors generally have two heat transfer systems, both of which typically contain sodium." The widely used term used by the SFR community is "sodium-cooled fast reactors." Why was "sodium-cooled fast reactors" not used throughout the report?

DOE Response:

There was no underlying technical reason for the use of sodium vs. sodium-cooled. SFR refers to a sodium-cooled fast reactor. The terms are equivalent.

40. General Question

In Section 3.1, Definitions, the DOE Report on page 7 specifies that structures, systems, and components (SSCs) that provide reasonable assurance the facility can be operated without undue risk to the health and safety of the public are designated as safety-related, and are relied upon to remain functional during design-basis accidents. However, Appendix A to 10 CFR Part 50 includes SSCs beyond only those that must remain functional during design-basis accidents. For example, SSCs that must be designed to not cause SSCs to fail to perform their safety functions because of seismic loads are included in Appendix A. Please comment on the consistency of the scope of the DOE Report with the current scope of SSCs covered by Appendix A to 10 CFR Part 50

DOE Response:

The scope of the DOE report is intended to be identical to that covered by the GDCs in 10 CFR 50, Appendix A. The words in the ARDCs that relate to protection of SSCs from hazards that could affect their function (such as the words in ARDCs 3 and 4 that discuss protection of SSCs from inadvertent actuation of the fire protection system and SSCs shall be appropriately protected against dynamic effects…) are identical to those in the GDCs.

The words on page 7 of the DOE report defining "Important to Safety" include the sentence, "SSCs with this designation are safety related and are relied upon to remain functional during design basis accidents." This sentence addresses the fact that safety-related SSCs must be protected from other hazards that could affect their function.

Appendix B

NRC Staff Questions on the DOE Report:

Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors
Department of Energy – Idaho National Laboratory
Docket No. PROJ 0814

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For Questions 1 - 40 see ADAMS ML15154B575

41. ARDC 17 and ARDC 18

As currently written, Title 10 Code of Federal Regulations (CFR) Part 50, Appendix A, General Design Criterion (GDC) 17, establishes electrical system requirements for a design-independent support system that can be applied to any reactor design. Among other things, GDC 17 requires that each plant have access to an offsite power electric system and an onsite electric power system. To inform its position with respect to non-light water reactors, the NRC staff would benefit from DOE's answers, comments and insights on the following questions:

a. Current GDC 17 establishes a set of power sources that a nuclear plant would utilize to meet its needs. If offsite power is considered fundamental to defense-in-depth, offsite power is a necessary element of electrical power systems requirements. Given that proposed Advanced Reactor Design Criterion (ARDC) 17 deletes offsite power from GDC 17, please explain why the DOE does not believe offsite power is a necessity for defense-in-depth for non-light water reactor designs.

b. If a new nuclear power plant design has connections to the offsite power system (grid), the design could be vulnerable to any disturbances or transients within the grid. For example, the onsite power system could be vulnerable to degraded grid voltage and loss-of-phase events. GDC 17 clarifies the NRC's requirements for specific protective measures to address these and other vulnerabilities and gives the NRC staff a firm regulatory basis to be able to enforce such requirements. GDC 18 outlines the inspection and testing requirements of electric power systems. How would the proposed ARDC 17 and 18 as currently worded provide the same level of protection and basis for enforcement of electrical systems?

42. HTGR-DC 33

In HTGR-DC 33, the following statement is made in the rationale, "...specified core radionuclide release design limits (SARRDLs) are not assured by the system addressed by this ARDC; adequate core cooling is maintained even with a depressurized primary circuit." Does this mean the SARRDLs are met with a depressurized circuit and hence no inventory control is necessary? Note in the first paragraph of the HTGR-DC 33 rationale discusses postulated accidents and not SARRDLs so it is unclear if inventory maintenance is needed for meeting the SARRDLs.

43. HTGR-DC 34

In HTGR-DC 34, the third paragraph of the rationale does not seem to correspond to any specific changes in the DC language. Please indicate which changes to the DC are made based on the third paragraph and provide additional discussion as to why the “design conditions of the reactor coolant pressure boundary” was deleted from the first paragraph of the DC.