PUBLIC WEBSITE – INVITATION AND INSTRUCTIONS FOR PUBLIC COMMENTS

Public Comment Sought - Advanced Non-Light Water Reactor Design Criteria

The NRC Regulatory Framework

In accordance with its mission, the U.S. Nuclear Regulatory Commission (NRC) protects the health and safety of the public and the environment by regulating the design, siting, construction, and operation of commercial nuclear power facilities. The NRC conducts its reactor licensing activities through a combination of regulatory requirements and regulatory guidance. The applicable regulatory requirements are found in Chapter I of Title 10, "Energy," of the Code of Federal Regulations (10 CFR). Chapter I is divided into Parts 1 through 199. Regulatory guidance is additional detailed information on specific acceptable means to meet the requirements in regulation. Guidance is provided in several forms such as in regulatory guides, interim staff guidance, standard review plans, office instructions, review standards, and Commission Policy Statements. These regulatory requirements and guidance represent the entirety of the regulatory framework that an applicant must consider when preparing an application for review by the NRC. A key part of the regulatory requirements is the “General Design Criteria for Nuclear Power Units,” which are contained in 10 CFR Part 50 Appendix A. The General Design Criteria (GDC) provide high-level requirements to support the design of nuclear power plants and are addressed in 10 CFR Part 50.34, “Contents of applications; technical information.” The current GDC are based on light water reactor technology. As discussed below, the attached non-light water reactor (non-LWR) design criteria were developed as guidance to more appropriately align with non-LWR technology. These non-LWR design criteria are the subject of this invitation for public comment.

The nuclear power plants presently operating in the United States were licensed under the process described in 10 CFR Part 50, Domestic Licensing of Production and Utilization Facilities.” The NRC and its predecessor, the Atomic Energy Commission, approved construction of these plants between 1964 and 1978 and granted the most recent operating license under 10 CFR Part 50 in 2015. 10 CFR Part 50 evolved over the years to address specific safety issues discovered as a result of operating experience and industry events. Some examples include fire protection in 10 CFR 50.48, emergency plans in 10 CFR 50.47, and aircraft impact assessment in 10 CFR 50.150. Some of these new regulations were applied retroactively to operating reactors while others applied only to new reactors.

The NRC applied its experience in licensing the currently operating fleet of nuclear power plants to the development of 10 CFR Part 52, “Licenses, Certifications, and Approvals for Nuclear Power Plants,” which was issued in 1989 and has been used for the most recent new nuclear power plant licensing reviews, reactor design certifications, and early site permits. The regulations in 10 CFR Part 52 are intended to apply lessons learned from licensing the current operating reactor fleet, provide an alternative licensing process to the licensing process described in 10 CFR Part 50, and increase standardization of the next generation of nuclear power plants. For many years, new nuclear power plant licensing and guidance development activities have focused on the licensing processes in 10 Part 52, rather than those in 10 CFR

Part 50. As a result, some Commission decisions regarding new nuclear power plant licensing issues have been incorporated into 10 CFR Part 52, without similar requirements consistently being incorporated into 10 CFR Part 50. For example, 10 CFR Part 52 includes requirements derived from the Commission "Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants" (ML003711521), with explicit requirements related to the Three Mile Island items in 10 CFR 50.34(f), severe accidents, probabilistic risk assessment, and other topics, whereas no similar requirements have been incorporated for new 10 CFR Part 50 nuclear power plant applications. In response to recent industry interest in employing the 10 CFR Part 50 process for new designs, SECY 15-0002, "Proposed Updates of Licensing Policies Rules, and Guidance for Future New Reactor Applications" (ML13277A647), was written to request that the Commission confirm that its policies and requirements apply to all new nuclear power plant applications, regardless of the selected licensing approach. The Commission approved the staff's recommendation that the regulations in 10 CFR Part 50 be revised for new power reactor applications to more closely align with requirements in 10 CFR Part 52.

Role of the General Design Criteria in the Regulatory Framework

As mentioned above, the GDC are contained in 10 CFR Part 50 Appendix A, and are an important part of the NRC's regulatory framework. They help to serve as the basis for design, fabrication, construction, testing, and performance requirements for structures, systems, and components (SSCs) that are important to safety; that is, as stated in Appendix A, "SSCs that provide reasonable assurance that the nuclear power plant can be operated without undue risk to the health and safety of the public." The GDC serve as the fundamental criteria for the NRC staff when reviewing the SSCs that make up a nuclear power plant design. They establish the design basis in that they address normal operations, anticipated operational occurrences and postulated accidents. As mentioned earlier, the regulatory framework includes the entire collection of regulation and guidance, which also address severe and beyond design basis accidents.

NRC Policy on Advanced Reactors

The NRC's mission with respect to regulating nuclear power reactors, consistent with its legislative mandate, is to ensure adequate protection of public health and safety, the common defense and security, and the environment. From the NRC staff's regulatory perspective, the characteristics of an "advanced reactor" has evolved over time, and this evolution is expected to continue. For example, the passive features in the AP1000 design were advanced concepts when first introduced. On October 14, 2008, the Commission issued its most recent policy statement regarding advanced reactors and included items to be considered during the design of such reactors. The Commission's 2008 "Policy Statement on the Regulation of Advanced Reactors" (ML082750370), reinforced and updated the policy statements regarding advanced reactors previously published in 1986 and 1994. In part, the 2008 update to the policy states the following:

> Regarding advanced reactors, the Commission expects, as a minimum, at least the same degree of protection of the environment and public health and safety and the common defense and security that is required for current generation light-water reactors [i.e., those licensed before 1997]. Furthermore, the Commission expects 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.

The Advanced Reactor Policy Statement makes clear the Commission's expectations that advanced reactor designs will address all current regulations including those related to severe accidents, beyond design basis accidents, defense-in-depth, and probabilistic risk assessment requirements. Depending on the design attributes of the different non-LWR technologies, regulations and policies may be addressed in different manner than traditional LWRs.

Role of the General Design Criteria for Advanced Non-LWRs

The requirements at 10 CFR 50.34(a)(3), 52.47(a)(3)(i), 52.79(a)(4), 52.137(a)(3) and 52.157(a) state that an application for a construction permit, design certification, combined license, standard design approval, or manufacturing license respectively, must include the principal design criteria (PDC) for the facility. The PDC are derived from the GDC in 10 CFR Part 50 Appendix A. 10 CFR Part 50, Appendix A establishes the applicability of the GDCs to non-LWR designs:

> These General Design Criteria establish minimum requirements for the principal design criteria for water-cooled nuclear power plants similar in design and location to plants for which construction permits have been issued by the Commission. The General Design Criteria are also considered to be generally applicable to other types of nuclear power units and are intended to provide guidance in establishing the principal design criteria for such other units.

In other words, the current regulations in 10 CFR Part 50, Appendix A, recognize that different requirements may be necessary for non-LWR designs. The preliminary draft of the advanced non-LWR design criteria as developed by the NRC staff are intended to provide stakeholders with insight into the staff's current views on how the General Design Criteria could be interpreted to address non-light water reactor design features; however, these are not considered to be final or binding regarding what may eventually be required from a non-LWR applicant. It is the applicant's responsibility to develop the PDC for its facility based on the specifics of its unique design, using the GDC, advanced non-LWR design criteria, or other design criteria as the foundation. Further, the applicant is responsible for considering public safety matters and fundamental concepts, such as defense in depth, in the design of their specific facility and for identifying and satisfying necessary safety requirements.

The advanced non-LWR design criteria are an important first step to address the unique characteristics of advanced non-LWR technology. Ultimately, a risk–informed, performance-based advanced non-LWR regulatory framework is envisioned. The NRC is open to new opportunities to explore a risk-informed performance-based regulatory process. The NRC recognizes the benefits to risk informing the advanced non-LWR design criteria to the extent possible, depending on the design information and data available.

DOE-NRC Initiative

In July 2013, the NRC and U.S. Department of Energy (DOE) established a joint initiative to address a key element in the regulatory framework that could apply to advanced, non-LWR technologies—specifically, addressing the existing GDC, which contain aspects that do not directly apply to non-LWR power plant designs. The purpose of the initiative is to assess the GDC to determine whether they apply to non-LWR designs and if not, to propose modifications to address the non-LWR design features. In each case, the underlying safety objective of the GDC still applies. These non-LWR design criteria are intended as regulatory guidance to assist the staff and future applicants. They are not regulatory requirements. 10 CFR Part 50.34(a)(3), "Contents of Applications; Technical Information," requires that an application for a design certification, combined license, standard design approval, or manufacturing license, include the principal design criteria (PDC) for a proposed facility. The non-LWR design criteria provide guidance intended to support the development of the PDC.

The assessment of the GDC with respect to non-LWR designs is being accomplished in two phases. Phase 1 was managed by a team including DOE representatives and its national laboratories, and consisted of reviews and evaluations of applicable technical information. The DOE team reviewed information related to six different types of non-light water reactor technologies (i.e., sodium-cooled fast reactors, lead fast reactors, gas-cooled fast reactors, modular high temperature gas-cooled reactors, fluoride high temperature reactors, and molten salt reactors). Using this information, the DOE then reviewed the existing NRC GDC to determine their applicability and whether they should be modified to reflect non-LWR designs.

The results of DOE's assessment are contained in a DOE report titled, "Guidance for Developing Principal Design Criteria for Advanced (Non-Light Water) Reactors." This report was submitted to the NRC for consideration in December 2014 and is publicly available (ML14353A246 and ML14353A248). In this report, DOE proposed a set of Advanced Reactor Design Criteria (ARDC), which could serve the same purpose for non-LWRs as the GDC serve for light water reactors. The ARDC are intended to be technology-neutral and, therefore, could potentially apply to any type of advanced non-LWR design.

In addition to the technology-neutral ARDC, DOE proposed two sets of technology-specific, non-LWR design criteria. These technology-specific design criteria are intended to apply to sodium fast reactors (SFRs) and modular high temperature gas reactors (mHTGRs), and are referred to as the SFR design criteria (SFR-DC) and the mHTGR design criteria (mHTGR-DC), respectively. During the review, the DOE determined that the safety objective for some of the current GDC were not applicable to SFR and mHTGR technologies so entirely new design criteria were developed to address unique design features (see section VIIa. and VIIb. of the NRC Draft Advanced Reactor Design Criteria Table).

The NRC is currently undertaking Phase 2 of the initiative. After receipt of the DOE report in December 2014, a multi-disciplinary team from across the NRC was assembled to review the report and other pertinent references and NRC documents, such as NUREGs, reports, and white papers. Some examples include NUREG-1338, “Pre-application Safety Evaluation Report for Modular High Temperature Gas-Cooled Reactor (mHTGR)” (ML052780497); NUREG-1368, “Pre-application Safety Evaluation Report for PRISM LMR” (ML063410561); and “Next Generation Nuclear Plant – Assessment of Key Licensing Issues” (ML14174A626). The NRC held a public meeting on January 21, 2015, (meeting summary available at ML15044A081) to discuss the report with DOE and to describe NRC’s plans to develop regulatory guidance for advanced reactor design criteria.

During its review, the NRC staff formulated questions and clarifications necessary to obtain a full understanding of design aspects of the non-LWR technologies and the reasoning that DOE employed in developing its proposal for the ARDC, SFR-DC, and mHTGR-DC. The NRC questions, and DOE responses to those questions, are publicly available at ADAMS Accession Numbers ML15154B575 and ML15223B331 (NRC letters), and ML15204A579 and ML15272A096 (DOE responses), respectively.

After consideration of the DOE report and other applicable information relevant to the NRC regulatory philosophy and current understanding of non-LWR designs, the NRC developed these draft safety ARDC, SFR-DC, and mHTGR-DC. It is important to note that the current GDC are regulations and therefore use the words “shall” and “must” that are appropriate for regulatory requirements. The proposed safety ARDC, SFR-DC, and mHTGR-DC also utilize the words “shall”, and “must” for consistency, but any Regulatory Guide that ultimately incorporates these design criteria will be guidance and not regulatory requirements. The “shall” and “must” language will apply only to those applicants that commit to the use of the Regulatory Guide. The NRC is not currently planning a rulemaking to add these advanced reactor design criteria to 10 CFR 50.

Process

The NRC staff believes that obtaining public comments on this draft version under development will be beneficial. Therefore, the ARDC, SFR-DC, and mHTGR-DC, along with the NRC’s initial rationale for each, are being made available on the NRC website for comment.

After receiving and considering comments, the NRC staff intends to develop a draft Regulatory Guide (RG) that will include revised ARDC, SFR-DC, and mHTGR-DC, as appropriate, and any related explanatory text. As part of the RG process, the draft RG will be made available for public comment through a federal register notice (FRN). After receiving and considering public comments on the draft RG, the NRC staff intends to issue a final RG that will provide guidance to non-LWR applicants when developing appropriate principal design criteria for their facilities.

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.

Other Advanced Non-LWR Activities

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.

The NRC is also considering a step-wise licensing strategy within the current NRC licensing framework in response to external stakeholders' expressed interest in finding an approach that will allow a potential applicant to address portions of a nuclear power plant design and applicable regulations as they are finalized. Agreed-upon portions of finalized design information would be submitted to gain regulatory feedback with the expectation that it is to support a future application. It is expected that proposed PDC for a non-LWR design will be a key early element to informing the content of future submittals.

Topics Open for Comment

The specific information on which the NRC is seeking comment is included in the Draft Advanced Reactor Design Criteria Table (Attachment 1). The table consists of eight sections (I –VII). The table in Sections I-VI has four columns. These ARDC, SFR-DC, and mHTGR-DC follow the existing GDC format:

Column 1 – Contains the current GDCs that are specified in 10 CFR Part 50, Appendix A. The NRC **is not seeking** comments on the information in this column because the requirements for light-water reactors are not being revised.

Column 2 - Contains the draft ARDC and the NRC's rationale for any adaptations from the current GDC. The NRC **is seeking** comments on the information in this column because this is new information.

Column 3 - Contains the draft SFR-DC and the NRC's rationale for the adaptations from the current GDC. The NRC **is seeking** comments on the information in this column because this is new information.

Column 4 - Contains the mHTGR-DC and the NRC's rationale for the adaptations from the GDC. The NRC **is seeking** comments on the information in this column because this is new information.

Section VII.a and VII.b contain additional SFR-DC and mHTGR-DC respectively. The NRC **is seeking** comments on the information in this column because this is new information.

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In addition to the contents of the columns described above, the NRC is specifically seeking comments on the following:

1. Are the ARDC generally applicable to the different types of non-LWRs being developed by different companies? Are there any additional criterion that should be added?
2. Should the current regulations that an applicant must address be incorporated into the ARDC? If so, which ones?
3. Are the SFR-DC and mHTGR-DC generally applicable to the different designs of SFRs and mHTGRs being developed by different companies? Are there any additional criterion that should be added?
4. There are several new approaches within the ARDC, SFR-DC, and mHTGR-DC, such as:
   - use of "functional containment" for mHTGR-DC,
   - use of "specified acceptable radionuclide release design limits" (SARRDLs) in the mHTGR-DC in place of specified acceptable fuel design limits (SAFDLs),
   - incorporation of GDC 35, "Emergency core cooling system," with GDC 34, "Residual heat removal," as applicable, and
   - the role of the SFR residual heat removal system during postulated accidents.

   Are these approaches appropriately addressed in the proposed criteria?

Commenting Instructions

Comments will be accepted for a 60 day period beginning on April 8, 2016, and ending June 8, 2016.

Comments can be made by using the Comments Form. Once you have completed entering your comments into the form, please click the "submit" button and the NRC will automatically receive your comments. Alternatively, you can email your comments to AdvancedRxDCComments.Resource@nrc.gov. Comments will not be responded to individually but will be considered by the NRC staff when developing the draft RG.

ATTACHMENT 1 - DRAFT Advanced Non-LWR Design Criteria Table – April 2016

	I. Overall Requirements			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
1	*Quality standards and records.* 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.	Same as GDC	Same as GDC	Same as GDC

ATTACHMENT 1 - DRAFT Advanced Non-LWR Design Criteria Table – April 2016

	I. Overall Requirements			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
2	*Design bases for protection against natural phenomena.* 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.	Same as GDC	Same as GDC	Same as GDC

	I. Overall Requirements			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
3	*Fire protection.* 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. 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.	*Fire protection.* 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. Rationale The phrase containing examples where noncombustible and heat	Same as ARDC	Same as ARDC

	I. Overall Requirements			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		resistant materials must be used has been broadened to apply to all advanced reactor designs.		

	I. Overall Requirements			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
4	*Environmental and dynamic effects design bases.* 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.	*Environmental and dynamic effects design bases.* 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.	Same as ARDC	Same as ARDC

	I. Overall Requirements			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		Rationale 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.		

	I. Overall Requirements			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
5	*Sharing of structures, systems, and components.* 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.	Same as GDC	Same as GDC	Same as GDC

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
10	*Reactor design.* 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.	Same as GDC	Same as GDC	*Reactor design.* 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. Rationale 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

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
				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

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
				and SARRDL are not exceeded. 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. 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 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. 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

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
				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.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
11	*Reactor inherent protection.* The reactor core and associated coolant systems 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.	*Reactor inherent protection.* 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. Rationale 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.	Same as ARDC	Same as ARDC

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
12	*Suppression of reactor power oscillations.* 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 design limits are not possible or can be reliably and readily detected and suppressed.	*Suppression of reactor power oscillations.* 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. Rationale 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.	Same as ARDC	*Suppression of reactor power oscillations.* 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. Rationale 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.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
13	*Instrumentation and control.* 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 systems within prescribed operating ranges	*Instrumentation and control.* 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 systems within prescribed operating ranges. Rationale "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	*Instrumentation and control.* 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~~ 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. Rationale "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.	*Instrumentation and control.* 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. Appropriate controls shall be provided to maintain these variables and systems within prescribed operating ranges. Rationale "As appropriate" was removed to provide specificity to the criterion. "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		to non-LWRs that operate at either low or high pressure.		The criterion has been modified to reflect use of the modular HTGR functional containment. See mHTGR-DC 16 rationale.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
14	*Reactor coolant pressure boundary.* 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.	*Reactor coolant ~~pressure~~ boundary.* 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. Rationale “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.	*Primary coolant ~~pressure~~ boundary.* 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. Rationale “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 only to the primary cooling system, not the intermediate cooling system. 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).	*Reactor helium~~coolant~~ pressure boundary.* 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. Rationale “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for mHTGRs. 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.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
15	*Reactor coolant system design.* 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.	*Reactor coolant system design.* 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. Rationale 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.	*~~Reactor~~ Primary coolant system design.* 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. Rationale “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” indicated that the SFR-DC is applicable only to the primary cooling system, not the intermediate cooling system. 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).	*Reactor helium pressure boundary ~~coolant system~~ design.* 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. Rationale “Reactor coolant system” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for mHTGRs.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
16	*Containment design.* 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.	Same as GDC Rationale For non-LWR technologies other than SFRs and mHTGRs, designers should use the current GDC to develop applicable principal design criteria.	*Containment design.* 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. 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. Rationale 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	*Containment design.* 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. Rationale The term "functional containment" is applicable to advanced non-LWRs without a pressure retaining containment structure. mHTGR-DC 16 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

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
			agreed that the containment leakage for advanced reactors, similar to and including PRISM, should not be required to meet the "essentially leaktight" statement in GDC 16. [Ref: NUREG-1368]. 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). 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.	long as postulated accident conditions require." 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. 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

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
				exclusion area boundary (EAB) with margins. The DOE report further clarifies functional containment in section 7.1.4: 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. 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

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
				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). 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 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).

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
17	*Electric power systems.* 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. 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. Electric power from the transmission network to the	*Electric power systems.* 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. 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. Electric power from the transmission network to the	*Electric power systems.* 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. 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. Electric power from the transmission network to the	*Electric power systems.* 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. 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.

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
	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 loss-of-coolant accident to assure that core cooling, containment integrity, and other vital safety functions are maintained. 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	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. 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	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. Provisions shall be included to minimize the probability of losing electric power from any of the remaining supplies as a	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. 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. Provisions shall be included to

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
	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.	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. Rationale 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. 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	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. Rationale 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. “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”	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. Rationale 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. “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		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. 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.	indicates that the SFR-DC is applicable to the primary cooling system, not the intermediate cooling system.	standard terms used for mHTGRs. 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.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
18	*Inspection and testing of electric power systems.* 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.	Same as GDC Rationale GDC 18 is a design-independent companion criterion to GDC 17.	Same as GDC Rationale GDC 18 is a design-independent companion criterion to GDC 17.	Same as GDC Rationale GDC 18 is a design-independent companion criterion to GDC 17.

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
19	*Control room.* 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 whole body, or its equivalent to any part of the body, for the duration of the accident. 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. Applicants for and holders of construction permits and operating licenses under this part who apply on or after	*Control room.* 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. Adequate habitability measures shall be provided to permit access and occupancy of the control room during normal operations and under accident conditions. 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	Same as ARDC	Same as ARDC

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
	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.	condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures. ~~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.~~		

	II. Multiple Barriers			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		Rationale The criterion was updated to remove specific emphasis on LOCA, which may be not appropriate for advanced designs such as the mHTGR. 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. 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. The last paragraph of the GDC has been eliminated for the ARDC because it is not applicable to future applicants.		

	II. Multiple Barriers			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
20	*Protection system functions.* 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.	Same as GDC Rationale For non-LWR technologies other than mHTGRs designers should use the current GDC to develop applicable principal design criteria.	Same as GDC	*Protection system functions.* 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 Rationale 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.

	III. Reactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
21	*Protection system reliability and testability.* 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.	Same as GDC	Same as GDC	Same as GDC

	III. Reactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
22	*Protection system independence.* 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.	Same as GDC	Same as GDC	Same as GDC

	III. Reactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
23	*Protection system failure modes.* 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.	Same as GDC Rationale For non-LWR technologies other than SFRs, designers should use the current GDC to develop applicable principal design criteria.	*Protection system failure modes.* 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. Rationale 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.	Same as GDC

	III. Reactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
24	*Separation of protection and control systems.* 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.	Same as GDC	Same as GDC	Same as GDC

	III. Reactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
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 for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.	*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 resulting from a ~~for any~~ single malfunction of the reactivity control systems. ~~, such as accidental withdrawal (not ejection or dropout) of control rods~~ Rationale 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.	Same as ARDC	*Protection system requirements for reactivity control malfunctions.* 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.~~ Rationale Use ARDC except SAFDL is replaced with SARRDL. The concept of using SARRDLs is discussed for GDC 10.

	III. Reactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
26	*Reactivity control system redundancy and capability.* 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.	*Reactivity control system redundancy and capability.* 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. Rationale "At least" was added to set a minimum number of	Same as ARDC	*Reactivity control system redundancy and capability.* 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 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. Rationale

	III. Reactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		independent reactivity control systems; it does not preclude more than two systems. 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. “Cold conditions” remains but will have to be defined by a principal design criteria for the specific design.		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.

	III. Reactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
27	*Combined reactivity control systems capability.* 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.	*Combined reactivity control systems capability.* 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. Rationale None of the advanced non-LWR designs evaluated in the review utilized poison addition via an ECCS. 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.	Same as ARDC	Same as ARDC

	III. Reactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
28	*Reactivity limits.* 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.	*Reactivity limits.* 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].~~ Rationale 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	*Reactivity limits.* 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].~~ Rationale “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	*Reactivity limits.* 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 dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition].~~ Rationale “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for mHTGRs. The list of “postulated reactivity accidents” has been

	III. Reactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		boundary" is applicable to non-LWRs that operate at either low or high pressure. The word "pressure" was deleted when referring to the reactor vessel as some designs may not be pressurized (SFR for example). 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.	cooling system, not the intermediate cooling system. 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.	deleted. Each design will have to determine its postulated reactivity accidents based on the specific design and associated risk evaluation.

	III. Reactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
29	*Protection against anticipated operational occurrences.* 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.	Same as GDC	Same as GDC	Same as GDC

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	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
30	*Quality of reactor coolant pressure boundary.* 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.	*Quality of reactor coolant ~~pressure~~ boundary.* 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. Rationale 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.	*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. Means shall be provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage. Rationale “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 only to the primary cooling system, not the intermediate cooling system. 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).	*Quality of reactor helium~~coolant~~ pressure boundary.* 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. Rationale “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for mHTGRs.

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
31	*Fracture prevention of reactor coolant pressure boundary.* 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.	*Fracture prevention of reactor coolant ~~pressure~~ boundary.* 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. Rationale 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	*Fracture prevention of ~~reactor~~ primary coolant ~~pressure~~ boundary.* 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. Rationale “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	*Fracture prevention of reactor helium ~~coolant~~ pressure boundary.* 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. Rationale “Reactor coolant pressure boundary” has been relabeled as “reactor helium pressure boundary” to conform to standard terms used for mHTGRs.

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
		such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure.	applicable only to the primary cooling system, not the intermediate cooling system. 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).	
	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
32	*Inspection of reactor coolant pressure boundary.* 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.	*Inspection of reactor coolant ~~pressure~~ boundary.* 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. Rationale 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	*Inspection of ~~reactor~~ primary coolant ~~pressure~~ boundary.* 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. Rationale "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	*Inspection of reactor helium~~coolant~~ pressure boundary.* 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. Rationale "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		such, the term "reactor coolant boundary" is applicable to non-LWRs that operate at either low or high pressure. 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.	applicable only to the primary cooling system, not the intermediate cooling system. 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). 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.	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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
33	*Reactor coolant makeup.* A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary shall be provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant 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.	*Reactor coolant inventory maintenance~~makeup~~.* 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.	*~~Reactor~~ Primary coolant inventory maintenance~~makeup~~.* 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.	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		Rationale Retitled with "inventory maintenance" to provide more flexibility regarding advanced reactor designs. 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. 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. 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	Rationale "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 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. 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.	

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		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.		

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
34	*Residual heat removal.* A system to remove residual heat shall be provided. The system safety function shall be to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded. 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.	*Residual heat removal.* 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. 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. 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	*Residual heat removal.* 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. 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. 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	*Passive residual heat removal.* 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. 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. 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

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		system safety function can be accomplished, assuming a single failure. Rationale ARDC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35. "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. 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. Text of first paragraph has been amended and the second paragraph added to clarify requirements that are applicable following normal operation including AOOs, and	power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure. 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. Rationale SFR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35. "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.	power is not available) the system safety function can be accomplished, assuming a single failure. Rationale mHTGR-DC 34 incorporates the postulated accident residual heat removal requirements contained in GDC 35. "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. 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. "Reactor coolant pressure boundary" has been relabeled as "reactor helium pressure boundary" to conform to standard terms used for mHTGRs. The specified acceptable core radionuclide release design

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		during postulated accidents following the precedent of NUREG-1368, "Pre-application SER for PRISM LMR." 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. 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	"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. 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. 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	limits replaces the ARDC specified acceptable fuel design limits as described in rationale to mHTGR-DC 10. 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		of power supply employed to address power reliability.	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.	

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
35	*Emergency core cooling.* 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 at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented and (2) clad metal-water reaction is limited to negligible amounts. 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.	*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 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. Rationale	Same as ARDC	Same as ARDC

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
		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.		

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
36	*Inspection of emergency core cooling system.* The emergency core cooling system 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.	*Inspection of ~~emergency core cooling~~ residual heat removal system.* 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. Rationale Title has been renamed and GDC revised to provide for inspection of the residual heat removal systems as required for ARDC 34.	Same as ARDC	*Inspection of passive ~~emergency core cooling~~ residual heat removal system.* 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. Rationale The word "passive" was added based on the definition of a mHTGR. In definitions Section 3.1 of INL/EXT-14-31179, the

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
		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. 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.		mHTGR design is defined as having passive heat removal due to a low power density. GDC 36 system is renamed and revised to provide for inspection of the residual heat removal systems as required for mHTGR-DC 34. Deleted the example list as they apply to LWR designs and each specific design will have different important components associated with residual heat removal.

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
37	*Testing of emergency core cooling system.* The 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 components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the	*Testing of residual heat removal ~~emergency core cooling~~ system.* 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	Same as ARDC	*Testing of passive residual heat removal ~~emergency core cooling~~ system.* 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

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
	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.	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. Rationale GDC 37 system has been renamed and revised to provide for testing of the residual heat removal system of ARDC 34. 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.		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 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. Rationale Criterion 37 has been renamed and revised for testing of the passive residual heat removal system required by modular HTGR-DC 34. 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

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		"Active" has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature. 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.		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. 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. 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
38	*Containment heat removal.* A system to remove heat from the reactor containment shall be provided. The system safety function shall be to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels. 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.	*Containment heat removal.* 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.~~ 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. Rationale "…as necessary…" is meant to condition ARDC 38 application to designs	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		requiring heat removal for conventional containments which are found to require heat removal measures. LOCA reference has been removed to provide for any postulated accident that might affect the containment structure. Containment structure safety system redundancy is addressed in second paragraph.		

ATTACHMENT 1 - DRAFT Advanced Non-LWR Design Criteria Table – April 2016

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
39	*Inspection of containment heat removal system.* 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.	*Inspection of containment heat removal system*. 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. Rationale Examples were deleted to make the ARDC technology neutral.	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
40	*Testing of containment heat removal system.* The containment 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 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.	*Testing of containment heat removal system.* 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. Rationale 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.	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
		"Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability. 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.		

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
41	*Containment atmosphere cleanup.* 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	*Containment atmosphere cleanup.* 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	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
	maintained. Each system shall have suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities 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) its safety function can be accomplished, assuming a single failure.	that containment integrity is maintained. 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. Rationale 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.		

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
42	*Inspection of containment atmosphere cleanup systems.* 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.	Same as GDC	Same as GDC	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
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 leaktight integrity of its components, (2) the operability and performance of the active 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, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.	*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 Rationale "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	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		examples of active system components. 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. 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. "Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability.		

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
44	*Cooling water.* A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink shall be provided. The system safety function shall be to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions. 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.	*Structural and equipment cooling ~~Cooling water~~*. 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. 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. Rationale This renamed ARDC accounts for advanced reactor design	Same as ARDC	*Structural and equipment cooling. ~~Cooling water.~~* 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. 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. Rationale Inserted "passive" based on system design for residual

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
		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.		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.
	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
45	*Inspection of cooling water system.* The cooling water system 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 system.	*Inspection of structural and equipment cooling ~~water~~* systems. 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. Rationale This renamed ARDC accounts for advanced reactor system design differences to include possible safety-related cooling required for SSCs.	Same as ARDC	Same as ARDC

	IV. Fluid Systems			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
46	*Testing of cooling water system.* The cooling water 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 the performance of the active 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 for reactor shutdown and for loss-of-coolant accidents, including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.	*Testing of structural and equipment cooling ~~. water~~ systems.* 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. Rationale This renamed ARDC accounts for advanced reactor system design differences to include	Same as ARDC	Same as ARDC

	IV. Fluid Systems			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		possible safety-related cooling required for SSCs. 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. "Leaktight" integrity would be demonstrated through appropriate functional testing of system performance and operability. "Active" has been deleted in item (2) as appropriate operability and performance system component testing is required regardless of active or passive nature. LOCA reference has been removed to provide for any postulated accident that might affect subject SSCs.		

	V. Reactor Containment			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
50	*Containment design basis.* 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 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 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.	*Containment design basis.* 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	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		the calculational model and input parameters. Rationale 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. 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. The example at the end of subpart 1 of the ARDC is LWR-specific and therefore deleted.		

	V. Reactor Containment			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
51	*Fracture prevention of containment pressure boundary.* The reactor containment boundary 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 of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.	*Fracture prevention of containment pressure boundary.* 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 of service temperatures and other conditions of the containment boundary materials during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws. Rationale 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	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		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. 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.		

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
52	*Capability for containment leakage rate testing.* The reactor containment 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.	*Capability for containment leakage rate testing*. 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. Rationale 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.	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
53	*Provisions for containment testing and inspection.* The reactor containment 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 leaktightness of penetrations which have resilient seals and expansion bellows.	*Provisions for containment testing and inspection.* 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. Rationale 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.	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	V. Reactor Containment			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
54	*Piping systems penetrating containment.* Piping systems penetrating primary reactor containment 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 piping systems shall be designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.	*Piping systems penetrating containment.* 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. Rationale 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	*Piping systems penetrating containment.* 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. Rationale 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	Same as ARDC Rationale 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 could indicate in its application that its specific mHTGR design makes this GDC not applicable.

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		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. 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.	make the phrase grammatically correct. 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 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. 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."	

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
			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. 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.	

	V. Reactor Containment			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
55	*Reactor coolant pressure boundary penetrating containment.* Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment 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: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (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 (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	*Reactor coolant ~~pressure~~ boundary penetrating containment.* 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 demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (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 (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	*~~Reactor~~ Primary coolant ~~pressure~~ boundary penetrating containment* 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: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (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 (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	Not applicable to modular HTGR. Rationale Lines that form a portion of the reactor coolant pressure boundary do not penetrate the reactor building. Therefore, this criterion does not apply.

	V. Reactor Containment			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
	valve outside containment. 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. 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.	valve outside containment. 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. 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.	outside containment. 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. 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.	

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		Rationale 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. 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.	Rationale 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. 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. 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	

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
			system and the steam generator in a PWR. "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.	

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
56	*Primary containment isolation.* Each line that connects directly to the containment atmosphere and penetrates primary reactor containment 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: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (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 (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. Isolation valves outside	~~*Primary*~~ *Containment isolation.* 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: (1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or (2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or (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 (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. Isolation valves outside	Same as ARDC	Not applicable to modular HTGR. Rationale 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.

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
	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.	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. Rationale 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.		

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
57	*Closed system isolation valves.* Each line that penetrates primary reactor containment 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 outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.	*Closed system isolation valves.* 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 outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve. Rationale 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.	*Closed system isolation valves.* 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. Rationale 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.	Not applicable to modular HTGR. Rationale 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.

	V. Reactor Containment			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
		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.	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.	

	VI. Fuel and Radioactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
60	*Control of releases of radioactive materials to the environment.* 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.	Same as GDC	Same as GDC	Same as GDC

	VI. Fuel and Radioactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
61	*Fuel storage and handling and radioactivity control.* 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 under accident conditions.	*Fuel storage and handling and radioactivity control.* 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. Rationale The underlying concept of establishing functional requirements for radioactivity control in fuel storage and fuel handling systems is independent of the design of	Same as ARDC	Same as ARDC

	VI. Fuel and Radioactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
		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.		

	VI. Fuel and Radioactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
62	*Prevention of criticality in fuel storage and handling.* Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.	Same as GDC	Same as GDC	Same as GDC

	VI. Fuel and Radioactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
63	*Monitoring fuel and waste storage.* 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.	Same as GDC	Same as GDC	Same as GDC

	VI. Fuel and Radioactivity Control			
Criterion	**Current GDC Language**	**ARDC Language/ Rationale for Modification**	**SFR-DC Language/ Rationale for Modification**	**mHTGR-DC Language/ Rationale for Modification**
64	*Monitoring radioactivity releases.* 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.	*Monitoring radioactivity releases.* 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. Rationale 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.	*Monitoring radioactivity releases.* 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. Rationale 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). Text was added to identify other SFR plant areas that should also be included to maintain consideration of all	*Monitoring radioactivity releases.* 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. Rationale 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.

	VI. Fuel and Radioactivity Control			
Criterion	Current GDC Language	ARDC Language/ Rationale for Modification	SFR-DC Language/ Rationale for Modification	mHTGR-DC Language/ Rationale for Modification
			potential discharge paths and 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.	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.

	VII.a. Additional SFR-DC
Criterion	SFR-DC Language/ Rationale for Modification
	Overarching Rationale for all Additional SFR-DC 10 CFR Part 50 Appendix A does not have a GDC corresponding to these SFR specific DC. NRC staff is considering the addition of SFR-DC 70 -77.
70	*Intermediate coolant system.* 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. Rationale 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).

	VII.a. Additional SFR-DC
Criterion	**SFR-DC Language/ Rationale for Modification**
71	*Primary coolant & cover gas purity control.* 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. Rationale 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) NUREG-1368 (page 3-57) (ML063410561) Section 3.2.4.6 suggested the 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.* 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. Rationale 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) 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).

	VII.a. Additional SFR-DC
Criterion	SFR-DC Language/ Rationale for Modification
73	*Sodium leakage detection and reaction prevention and mitigation*. 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. Rationale 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 Advanced Non-Light Water Reactors" (ML15204A579) (pages 15-16). 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).
74	*Sodium/water reaction prevention/mitigation.* 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. 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. Rationale 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.

	VII.a. Additional SFR-DC
Criterion	**SFR-DC Language/ Rationale for Modification**
75	*Quality of the intermediate coolant boundary.* 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. Rationale 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.
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. Rationale 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.
77	*Inspection of the intermediate coolant boundary.* 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. Rationale 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.

	VII.b. Additional mHTGR-DC
Criterion	**mHTGR-DC Language/ Rationale for Modification**
	Overarching Rationale for all Additional mHTGR-DC 10 CFR Part 50 Appendix A does not have a GDC corresponding to this mHTGR specific DC. NRC staff is considering the addition of mHTGR-DC 70-72.
70	*Reactor vessel and reactor system structural design basis.* 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. Rationale 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.
71	*Reactor building design basis.* 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. Rationale 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. 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).
72	*Provisions for periodic reactor building inspection*. 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. Rationale This newly established criterion regarding periodic inspection and surveillance provides assurance that the reactor building will perform its safety functions of protecting and maintaining the configuration needed for passive cooling and providing a discharge pathway for helium depressurization events.